August 19, 1986

Docket No.: 50-260

Mr. S. A. White
Manager of Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37401

Dear Mr. White:

The Commission has issued the enclosed Amendment No. 125 , to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit 1. This amendment is in response to your application dated August 23, 1984 (TVA BFNP TS-199), as supplemented September 4 and November 13, 1984, April 3, May 8, June 27, November 20 and December 30, 1985 and April 29, 1986.

The amendment revises the Technical Specifications (TS) of the operating license to: (1) modify the core physics, thermal and hydraulic limits to be consistent with the reanalyses associated with replacing about one-third of the core during the Cycle 6 core reload outage and (2) reflect changes in various specifications as a result of plant modifications performed during the outage. In addition, TVA has updated the TS pages involved and made administrative corrections.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Marshall Grotenhuis, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Use as Original

Veronica 8/26/86

Enclosures:
1. Amendment No. 125 to License No. DPR-52
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:
Docket File
NRC PDR
Local PDR
JPartlow
LHarmon
CStahle
SRConnelly,
BHayes,OI
HDenton
RBernero
SNorris
MGrotenhuis
OGC
ACRS (10)
FCantrell,RII
TKenyon
GZech,RII
NGrace,RII
HThompson
EJordan
BGrimes
RClark
WJones
RWessman
WLong
BJYoungblood
LSpessard,DI
SRichardson,IE
DVassallo
OPA
TBarnhart (4)
Plant File
DMuller
TAlexion
JHolonich
SWeise,RII
JTaylor,IE
HThompson
LFMB

8608280246 860819
PDR ADOCK 05000260
P PDR

OFFICIAL RECORD COPY

DBL:PD#2	DBL:PD#2	DBL:PD#2	OGC	DBL:PD#2:D
SNorris	MGrotenhuis	RClark	J. Goldberg	DMuller
08/12/86	08/11/86	08/11/86	08/13/86	08/19/86

cc:

H. S. Sanger, Jr., Esquire
General Counsel
Tennessee Valley Authority
400 Commerce Avenue
E 11B 330
Knoxville, Tennessee 37902

Mr. Ron Rogers
Tennessee Valley Authority
5N 130B Lookout Place
Chattanooga, Tennessee 37402-2801

Chairman, Limestone County Commission
Post Office Box 188
Athens, Alabama 35611

Ira L. Meyers, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36130

Mr. K. W. Whitt
E3A8
400 West Summit Hill Drive
Tennessee Valley Authority
Knoxville, Tennessee 37902

Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Mr. Steven Roessler
U. S. Nuclear Regulatory Commission
Reactor Training Center
Osborne Office Center, Suite 200
Chattanooga, Tennessee 37411

W. C. Bibb
Site Director, BFNP
Tennessee Valley Authority
Post Office Box 2000
Decatur, Alabama 35602

Resident Inspector
U. S. Nuclear Regulatory Commission
Route 2, Box 311
Athens, Alabama 35611

Mr. Donald L. Williams, Jr.
Tennessee Valley Authority
400 West Summit Hill Drive, W10B85
Knoxville, Tennessee 37902

Robert L. Lewis, Manager, BFNP
Tennessee Valley Authority
Post Office Box 2000
Decatur, Alabama 35602

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 125
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated August 23, 1984 (TVA BFNP TS-199), as supplemented September 4 and November 13, 1984, April 3, May 8, June 27, November 20 and December 30, 1985 and April 29, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility Operating License No. DPR-52 is hereby amended to read as follows:

(2) Technical Specification

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 125 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as the date of its issuance and is to be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 19, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 125

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

1. Remove the following pages and replace with identically numbered pages.

2. The marginal lines on these pages denote the area being changed.

iv	62	250
vi	63	256
vii	73	257
viii	78	262
3	79	263
4	80	330
9	85	356
19	96	
23	102	
25	105	
28	105a	
32	110	
33	110a**	
34	159	
35	160	
37	168a	
38	169	
39	171	
40	172	
41	172a*	
42	220	
44	231	
55	232	

*Page 172a is removed but there is no replacement.
**Page Added

LIST OF TABLES

LIST OF TABLES (Cont'd)

Amendment No. 125

LIST OF ILLUSTRATIONS

Amendment No. ~~85~~ ,125

1.0 DEFINITIONS (cont'd)

E. Operable - Operability - A system, subsystem, train, component, or device shall be Operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

F. Operating - Operating means that a system or component is performing its intended functions in its required manner.

G. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

H. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup" or "Run" position with the reactor critical and above 1% rated power.

I. Hot Standby Condition - Hot standby condition means operation with coolant temperature greater than 212°F, system pressure less than 1055 psig, the main steam isolation valves closed and the mode switch in the Startup/Hot Standby position.

J. Cold Condition - Reactor coolant temperature equal to or less than 212°F.

K. Hot Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature greater than 212°F.

L. Cold Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature equal to or less than 212°F.

M. Mode of Operation - A reactor mode switch selects the proper interlocks for the operational status of the unit. The following are the modes and interlocks provided:

1. Startup/Hot Standby Mode - In this mode the reactor protection system is energized with IRM neutron monitoring system trip, the APRM 15% high flux trip, and control rod withdrawal interlocks in service. This is often referred to as just Startup Mode. This is intended to imply the startup/Hot Standby position of the mode switch.

1.0 DEFINITIONS (Cont'd)

2. Run Mode - In this mode the reactor system pressure is at or above 825 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and the RBM interlocks in service.

3. Shutdown Mode - Placing the mode switch to the shutdown position initiates a reactor scram and power to the control rod drives is removed. After a short time period (about 10 sec), the scram signal is removed allowing a scram reset and restoring the normal valve lineup in the control rod drive hydraulic system.

4. Refuel Mode - With the mode switch in the refuel position interlocks are established so that one control rod only may be withdrawn when the Source Range Monitor indicate at least 3 cps and the refueling crane is not over the reactor except as specified by TS 3.10.B.1.b.2. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

N. Rated Power - Rated power refers to operation at a reactor power of 3,293 MWt; this is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power. Design power, the power to which the safety analysis applies, corresponds to 3,440 MWt.

O. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All non-automatic containment isolation valves on lines connected to the reactor coolant systems or containment which are not required to be open during accident conditions are closed. These valves may be opened to perform necessary operational activities.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

1.1 FUEL CLADDING INTEGRITY

2.1 FUEL CLADDING INTEGRITY

b.

c. 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 $\leq$13.4 kw/ft and MCPR within limits of Specification 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.

d. The APRM Rod block trip setting shall be:

$S_{RB} \leq (0.66W + 42\%)$

where:

S_{RB} = Rod block setting in percent of rated thermal power (3293 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

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 comparisons and results are summarized in Reference 1.

The void reactivity coefficient and the scram worth are described in detail in reference 1.

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 1. 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 BASIS

from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system.

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

The set point 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 approximately 31 inches below the normal operating range and is thus adequate 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% 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% 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. Relevant transient analyses are discussed in References 1 and 2. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first state pressure.

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

2. Generic Reload Fuel Application, Licensing Topical Report NEDE-20411-P-A, and Addenda.

Amendment No. ~~85~~, 125

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 pressure (1,148 psig for suction and 1,326 psig 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 are 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 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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.1 REACTOR PROTECTION SYSTEM	4.1 REACTOR PROTECTION SYSTEM
B. Two RPS power monitoring channels for each inservice RPS MG sets or alternate source shall be operable.	**B. The RPS power monitoring system instrumentation shall be determined operable:** **At least once per 6 months by performance of channel functional tests.**
1. With one RPS electric power monitoring channel for inservice RPS MG set or alternate power supply inoperable, restore the inoperable channel to operable status within 72 hours or remove the associated RPS MG set or alternate power supply from service.	
2 With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one to operable status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.	

Amendment No. 125

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Min. No. of Operable Inst. Channels Per Trip System(1)(23)	Trip Function	Trip Level Setting	Shut-down	Modes in Which Function Must Be Operable: Refuel(7)	Startup/Hot Standby	Run	Action(1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤120/125 Indicated on scale	X(22)	X(22)	X	(5)	1.A
3	Inoperable			X	X	(5)	1.A
	APRM (16) (24)(25)						
2	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B
2	High Flux (Fixed Trip)	≤120 %				X	1.A or 1.B
2	High Flux	≤15% rated power		X(21)	X(17)	(15)	1.A or 1.B
2	Inoperative	(13)		X(21)	X(17)	X	1.A or 1.B
2	Downscale	≥3 Indicated on Scale		(11)	(11)	X(12)	1.A or 1.B
2	High Reactor Pressure (PIS-3-22AA, BB, C, D)	≤1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14) (PIS-64-56 A-D)	≤2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14) (LIS-3-203 A-D)	≥538" above vessel zero		X	X	X	1.A
2	High Water Level in West Scram Discharge Tank (LS-85-45 A-D)	≤50 Gallons	X	X(2)	X	X	1.A
2	High Water Level in East Scram Discharge Tank (LS-85-45E-H)	≤50 Gallons	X	X(2)	X	X	1.A

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Min. No. of Operable Inst. Channels Per Trip System(1) (23)	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable				Action(1)
			Shut-down	Refuel(7)	Startup/Hot Standby	Run	
4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure				X(6)	1.A or 1.C
2	Turbine Cont. Valve Fast Closure or Turbine Trip	≥550 psig				X(4)	1.A or 1.D
4	Turbine Stop Valve Closure	≤10% Valve Closure				X(4)	1.A or 1.D
2	Turbine First Stage Pressure Permissive (PIS-1-81A&B, PIS-1-91A&B)	not ≥154 psig		X(18)	X(18)	X(18)	(19)
2	Main Steam Line High Radiation (14)	3X Normal Full Power Background (20)		X(9)	X(9)	X(9)	1.A or 1.C
2	Low Scram Pilot Air Header Pressure	≥50 psig	X(2)	X(2)	X	X	1.A

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable instrument channels per trip system cannot be met for one trip system, trip the inoperable channels or entire trip system within one hour, or, alternatively, take the below listed action for that trip function. If the minimum number of operable instrument channels cannot be met by either trip system, the appropriate action listed below (refer to right-hand column of Table) shall be taken. An inoperable channel need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the inoperable channel shall be restored to operable status within two hours, or take the action listed below for that trip function.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours. In refueling mode, suspend all operations involving core alterations and fully insert all operable control rods within one hour.

   B. Reduce power level to IRM range and place mode switch in the Startup|Hot Standby position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30% of rated.

2. Scram discharge volume high bypass may be used in shutdown or refuel to bypass scram discharge volume scram and scram pilot air header low pressure scram with control rod block for reactor protection system reset.

3. DELETED.

4. Bypassed when turbine first stage pressure is less than 154 psig.

5. IRMs are bypassed when APRMs are onscale and the reactor mode switch is in the run position.

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212°F. only the following trip functions need to be operable:

   A. Mode switch in shutdown
   B. Manual scram
   C. High flux IRM
   D. Scram discharge volume high level
   E. APRM 15% scram
   F. Scram pilot air header low pressure

8. Not required to be operable when primary containment integrity is not required.

9. Not required if all main steamlines are isolated.

3

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

	Group (2)	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once Per Week During Refuelin and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once Per Week During Refuelin and Before Each Startup
APRM			
High Flux (15% scram)	C	Trip Output Relays (4)	Before Each Startup and Weekl When Required to be Operable
High Flux (Flow Biased)	B	Trip Output Relays (4)	Once/Week
High Flux (Fixed Trip)	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	(6)	(6)
High Reactor Pressure (PIS-3-22AA, BB, C, D)	B	Trip Channel and Alarm (7)	Once/ month
High Drywell Pressure (PIS-64-56 A-D)	B	Trip Channel and [illegible]larm (7)	Once/ month
Reactor Low Water Level (LIS-3-203 A-D)	B	Trip Channel and Alarm (7)	Once/ month
High Water Level in Scram Discharge Tank			
Float Switches (LS-85-45 C-F)	A	Trip Channel and Alarm	Once/month
Electronic Level Switches (LS-85-45A, B, G, H)	B	Trip Channel and Alarm (7)	Once/ month
Main Steam Line High Radiation	B	Trip Channel and Alarm (4)	Once/3 months (8)

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

	Group (2)	Functional Test	Minimum Frequency (3)
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Once/3 Months (8)
Turbine Control Valve Fast Closure or Turbine Trip	A	Trip Channel and Alarm	Once/Month (1)
Turbine First Stage Pressure Permissive (PIS-1-81 A&B, PIS-1-91 A&B)	B	Trip Channel and Alarm (7)	Every 3 Months.
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Once/Month (1)
Low Scram Pilot Air Header Pressure PS 85-35 A1, A2, B1, & B2	A	Trip Channel and Alarm	Once/6 Months

NOTES FOR TABLE 4.1.A

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

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

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

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

5. (DELETED)

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

7. Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify operability of the trip and alarm functions.

8. The functional test frequency decreased to once/3 months to reduce challenges to relief valves per NUREG 0737, Item II.K.3.16.

TABLE 4.1.B
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled Startups (6)	Note (4)
APRM High Flux			
Output Signal	B	Heat Balance	Once every 7 days
Flow Bias Signal	B	Calibrate Flow Bias Signal (7)	Once/operating cycle
LPRM Signal	B	TIP System Traverse (8)	Every 1000 Effective Full Power Hours
High Reactor Pressure (PIS-3-22AA, BB, C, D)	B	Standard Pressure Source	Once/18 Months (9)
High Drywell Pressure (PIS-64-56 A-D)	B	Standard Pressure Source	Once/18 Months (9)
Reactor Low Water Level (LIS-3-203 A-D)	B	Pressure Standard	Once/18 Months (9)
High Water Level in Scram Discharge Volume Float Switches (LS-85-45 C-F)	A	Calibrated Water Column	Once/18 Months
Electronic Level Switches (LS-85-45 A, B, G, H)	B	Calibrated Water Column	Once/18 Months (9)
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 Months
Turbine First Stage Pressure Permissive (PIS-1-81 A&B, PIS-1-91 A&B)	B	Standard Pressure Source	Once/18 Months (9)
Turbine Stop Valve Closure	A	Note (5)	Note (5)
Turbine Cont. Valve Fast Closure on Turbine Trip	A	Standard Pressure Source	Once/Operating Cycle
Low Scram Pilot Air Header Pressure PS 85-35 A1, A2, B1 & B2	A	Standard Pressure Source	Once/18 Months

NOTES FOR TABLE 4.1.B

1. A description of three groups is included in the bases of this specification.

2. Calibrations are not required when the systems are not required to be operable or are tripped. If calibrations are missed, they shall be performed prior to returning the system to an operable status.

3. The current source provides an instrument channel alignment. Calibration using a radiation source shall be made each refueling outage.

4. Required frequency is initial startup following each refueling outage.

5. Physical inspection and actuation of these position switches will be performed once per operating cycle.

6. On controlled startups , overlap between the IRM's and APRM's will be verified.

7. The Flow Bias Signal Calibration will consist of calibrating the sensors, flow converters, and signal offset networks during each operating cycle. The instrumentation is an analog type with redundant flow signals that can be compared. The flow comparator trip and upscale will be functionally tested according to Table 4.2.C to ensure the proper operating during the operating cycle. Refer to 4.1 Bases for further explanation of calibration frequency.

8. A complete tip system traverse calibrates the LPRM signals to the process computer. The individual LPRM meter readings will be adjusted as a minimum at the beginning of each operating cycle before reaching 100% power.

9. Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the trip level setting.

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 non-class IE power supply and the class IE RPS bus. This will ensure that failure of a non-class IE reactor protection power supply will not cause adverse interaction to the class IE 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.

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 Specification 2.1 and 2.2.

modes. In the power range the APRM system provides required protection. Ref. Section 7.5.7 FSAR. Thus, the IRM System is not required in the Run mode. The APRM's and the IRM's 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.

The requirement to have the scram functions as indicated in Table 3.1.A operable in the Refuel mode is to assure that shifting to the Refuel mode during reactor power operation does not diminish the need for the reactor protection system.

Because of the APRM downscale limit of $\geq$ 3% when in the Run mode and high level limit of $\leq$ 15% when in the Startup Mode, the transition between the Startup and Run Modes must be made with the APRM instrumentation indicating between 3% and 15% 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 IRM's 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.

Amendment Nos. 38,49,83,103,108,125

TABLE 3.2.A

PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level (6) (LIS-3-203 A-D)	≥ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure	100 ± 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56 A-D)	≥ 470" above vessel zero	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PIS-64-56 A-D)	≤ 2.5 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
2	Instrument Channel - High Radiation Main Steam Line Tunnel (6)	≤ 3 times normal rated full power background	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Low Pressure Main Steam Line (PIS-1-72, 76, 82, 86)	≥ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel - High Flow Main Steam Line (PdIS-1-13A-D, 25A-D, 36A-D, 50A-D)	≤ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation

55

TABLE 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Below trip setting initiated HPCI.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 378" above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 378" above vessel zero.	A	1. Below trip settings in conjunction with drywell high pressure, low water level permissive, 120 sec. delay timer and CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184, 185)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LIS-3-52, 62)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.
2	Instrument Channel - Drywell High Pressure (PIS-64-58E-H)	1≤ P≤2.5 psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.

Amendment Nos. 38,62,125

Amendment Nos. 38,62,125

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Drywell High Pressure (PIS-64-58A-D)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal. (15)
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D)	≥ 470" above vessel zero	A	1. Below trip setting trips recirculation pumps
2	Instrument Channel Reactor High Pressure (PIS-3-204A-D)	≤ 1120 psig	A	1. Above trip setting trips recirculation pumps
2	Instrument Channel - Drywell High Pressure (PIS-64-58A-D)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
2(16)	Instrument Channel - Drywell High Pressure (PIS-64-57 A-D)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor water level, drywell high pressure, 120 sec. delay timer and CSS or RHR pump running, initiates ADS.
2	Instrument Channel - Reactor Low Pressure (PIS-3-74A&B) (PIS-68-95, 96)	450 psig ± 15	A	1. Below trip setting permissive for opening CSS and LPCI admission valves.
2	Instrument Channel - Reactor Low Pressure (PS-3-74A&B) (PS-68-95, 96)	230 psig ± 15	A	1. Recirculation discharge valve actuation.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5)	Function	Trip Level Setting
4(1)	APRM Upscale (Flow Bias)	$\leq$0.66W + 42% (2)
4(1)	APRM Upscale (Startup Mode) (8)	$\leq$12%
4(1)	APRM Downscale (9)	$\geq$3%
4(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (Flow Bias)	$\leq$0.66W + 40% (2)(13)
2(7)	RBM Downscale (9)	$\geq$3%
2(7)	RBM Inoperative	(10c)
6(1)	IRM Upscale (8)	$\leq$108/125 of full scale
6(1)	IRM Downscale (3) (8)	$\geq$5/125 of full scale
6(1)	IRM Detector not in Startup Position (8)	(11)
6(1)	IRM Inoperative (8)	(10a)
3(1) (6)	SRM Upscale (8)	$\leq$ $1X10^5$ counts/sec.
3(1) (6)	SRM Downscale (4) (8)	$\geq$3 counts/sec.
3(1) (6)	SRM Detector not in Startup Position (4)(8)	(11)
3(1) (6)	SRM Inoperative (8)	(10a)
2(1)	Flow Bias Comparator	$\leq$10% difference in recirculation flows
2(1)	Flow Bias Upscale	$\leq$115% recirculation flow
1	Rod Block Logic	N/A
2(1)	RCSC Restraint (PS85-61A,B)	147 psig turbine first stage pressure
1(12)	High Water Level in West Scram Discharge Tank (LS-85-45L)	$\leq$ 25 gal.
1(12)	High Water Level in East Scram Discharge Tank (LS-85-45M)	$\leq$ 25 gal.

Amendment Nos. 67,87,107,125

TABLE 3.2.F
SURVEILLANCE INSTRUMENTATION

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

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) (3)
1/Valve		Relief Valve Tailpipe Thermocouple Temperature or Acoustic Monitor on Relief Valve Tailpipe		(5)
1	RR-90-272CD RR-90-273CD	High Range Primary Containment Radiation Recorders	Recorder, 1 - 10 R/Hr	(7) (8)
2	LI-64-159A XR-64-159	Suppression Chamber Water Level-Wide Range	Indicator, Recorder 0-240"	(1) (2) (3)
2	PI-64- 160A XR-64-159	Drywell Pressure Wide Range	Indicator, Recorder 0-300 psig	(1) (2) (3)
2	TI-64-161 TR-64-161 TI-64-162 TR-64-162	Suppression Pool Bulk Temperature	Indicator, Recorder 30° - 230° F	(1) (2) (3) (4) (6)
1	RR-90-322A	Wide Range Gaseous Effluent Radiation Monitor	Recorder Noble Gas) 10^{-7} - 10^{+5} µCi/cc) Iodine and Particulates) 10^{-12} - 10^{+2} µCi/cc)	(7) (8)

NOTES FOR TABLE 3.2.F

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

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

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

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

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

(6) A channel consists of 8 sensors, one from each alternating torus bay. Seven sensors must be operable for the channel to be operable.

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

(8) With the plant in the power operation, startup, or hot shutdown condition and with the number of operable channels less than the required operable channels, either restore the inoperable channel(s) to operable status within 72 hours, or initiate the preplanned alternate method of monitoring the appropriate parameter.

Amendment Nos. ~~63~~, ~~68~~, 125

TABLE 4.2.A

SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency		Instrument Check
Instrument Channel - Reactor Low Water Level (LIS-3-203A-D)	(1) (27)	Once/18 Months	(28)	Once/day
Instrument Channel - Reactor High Pressure	(1)	Once/3 Months		None
Instrument Channel - Reactor Low Water Level (LIS-3-56A-D)	(1) (27)	Once/18 Months	(28)	Once/day
Instrument Channel - High Drywell Pressure (PIS-64-56A-D)	(1) (27)	Once/18 Months	(28)	N/A
Instrument Channel - High Radiation Main Steam Line Tunnel	(29)	(5)		Once/day
Instrument Channel - Low Pressure Main Steam Line (PIS-1-72, 76, 82, 86)	(29) (27)	Once/18 Months	(28)	None
Instrument Channel - High Flow Main Steam Line (PdIS-1-13A-D, 25A-D, 36A-D, 50A-D)	(29) (27)	Once/18 Months	(28)	Once/day
Instrument Channel - Main Steam Line Tunnel High Temperature	(29)	Once/operating cycle		None
Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	(1)(14)(22)	Once/3 Months	[redacted]	Once/day(0)

Amendment No. 105,125

TABLE 4.2.D
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
Instrument Channel Reactor Low Water Level (LIS-3-58A-D)	(1) (27)	Once/18 Months (28)	Once/day
Instrument Channel Reactor Low Water Level (LIS-3-184 & 185)	(1) (27)	Once/18 Months (28)	Once/day
Instrument Channel Reactor Low Water Level (LIS-3-52 & 62)	(1) (27)	Once/18 Months (28)	Once/day
Instrument Channel Reactor Low Water Level (LIS-3-56A-D)	(1) (27)	Once/18 Months (28)	None
Instrument Channel Reactor High Pressure (PIS-3-204A-D)	(1) (27)	Once/18 Months (28)	None
Instrument Channel Drywell High Pressure (PIS-64-58E-H)	(1) (27)	Once/18 Months (28)	None
Instrument Channel Drywell High Pressure (PIS-64-58A-D)	(1) (27)	Once/18 Months (28)	None
Instrument Channel Drywell High Pressure (PIS-64-57A-D)	(1) (27)	Once/18 Months (28)	None
Instrument Channel Reactor Low Pressure (PIS-3-74A&B, PS-3-74A&B) (PIS-68-95, PS-68-95) (PIS-68-96, PS-68-96)	(1) (27)	Once/18 Months (28)	None

TABLE 4.2.C

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE ROD BLOCKS

Function	Functional Test	Calibration (17)	Instrument Check
APRM Upscale (Flow Bias)	(1) (13)	Once/3 Months	Once/day(8)
APRM Upscale (Startup Mode)	(1) (13)	Once/3 Months	Once/day(8)
APRM Downscale	(1) (13)	Once/3 Months	Once/day(8)
APRM Inoperative	(1) (13)	N/A	Once/day(8)
RBM Upscale (Flow Bias)	(1) (13)	Once/6 Months	Once/day(8)
RBM Downscale	(1) (13)	Once/6 Months	Once/day(8)
RBM Inoperative	(1) (13)	N/A	Once/day(8)
IRM Upscale	(1) (2) (13)	Once/3 Months	Once/day(8)
IRM Downscale	(1) (2) (13)	Once/3 Months	Once/day(8)
IRM Detector not in Startup Position	(2) (Once/operating cycle)	Once/operating cycle (12)	N/A
IRM Inoperative	(1) (2) (13)	N/A	N/A
SRM Upscale	(1) (2) (13)	Once/3 Months	Once/day(8)
SRM Downscale	(1) (2) (13)	Once/3 Months	Once/day(8)
SRM Detector not in Startup Position	(2) (Once/operating cycle)	Once/operating cycle (12)	N/A
SRM Inoperative	(1) (2) (13)	N/A	N/A
Flow Bias Comparator	(1) (15)	Once/operating cycle (20)	N/A
Flow Bias Upscale	(1) (15)	Once/3 Months	N/A
Rod Block Logic	(16)	N/A	N/A
RSCS Restraint	(1)	Once/3 Months	N/A
West Scram Discharge Tank Water Level High (LS-85-45L)	Once/quarter	Once/18 Months	N/A
East Scram Discharge Tank Water Level High (LS-85-45M)	Once/quarter	Once/18 Months	N/A

Amendment Nos. 38,42 125

TABLE 4.2.F
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

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

TABLE 4.2.F
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
17 Relief valve Tailpipe Thermocouple Temperature	NA	Once/month (24)
18 Acoustic Monitor on Relief Valve Tailpipe	Once/cycle (25)	Once/month (26)
19 High-Range Primary Containment Radiation Monitors (RR-90-272CD) (RR-90-273CD)	Once/18 months (30)	Once/month
20 Suppression Chamber Water Level-Wide Range (LI-64-159A) (XR-64-159)	Once/18 months	Once/month
21 Drywell Pressure-Wide Range (PI-64-160A) (XR-64-159)	Once/18 months	Once/shift
22 Suppression Pool Bulk Temperature (TI-64-161) (TR-64-161) (TI-64-162) (TR-64-162)	Once/18 months	Once/shift
23 High Range Gaseous Effluent Radiation Monitor (RR-90-322A)	Once/18 months	Once/shift

NOTES FOR TABLES 4.2.A THROUGH 4.2.H (Continued)

14. Upscale trip is functionally tested during functional test time as required by section 4.7.B.1.a and 4.7.C.1.c.

15. The flow bias comparator will be tested by putting one flow unit in "Test" (producing 1/2 scram) and adjusting the test input to obtain comparator rod block. The flow bias upscale will be verified by observing a local upscale trip light during operation and verified that it will produce a rod block during the operating cycle.

16. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

17. This calibration consists of removing the function from service and performing an electronic calibration of the channel.

18. Functional test is limited to the condition where secondary containment integrity is not required as specified in sections 3.7.C.2 and 3.7.C.3.

19. Functional test is limited to the time where the SGTS is required to meet the requirements of section 4.7.C.1.a.

20. Calibration of the comparator requires the inputs from both recirculation loops to be interrupted, thereby removing the flow bias signal to the APRM and RBM and scramming the reactor. This calibration can only be performed during an outage.

21. Logic test is limited to the time where actual operation of the equipment is permissible.

22. One channel of either the reactor zone or refueling zone Reactor Building Ventilation Radiation Monitoring System may be administratively bypassed for a period not to exceed 24 hours for functional testing and calibration.

23. (Deleted)

24. This instrument check consists of comparing the thermocouple readings for all valves for consistence and for nominal expected values (not required during refueling outages).

25. During each refueling outage, all acoustic monitoring channels shall be calibrated. This calibration includes verification of accelerometer response due to mechanical excitation in the vicinity of the sensor.

26. This instrument check consists of comparing the background signal levels for all valves for consistency and for nominal expected values (not required during refueling outages).

NOTES FOR TABLES 4.2.A THROUGH 4.2.H (Continued)

27. Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify operability of the trip and alarm functions.

28. Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the trip level setting.

29. The functional test frequency decreased to once/3 months to reduce challenges to relief valves per NUREG-0737, Item II.K.3.16.

30. Calibration shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one-point source check of the detector below 10 R/hr with an installed or portable gamma source.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.H **Maintenance of Filled Discharge Pipe**	4.5.H **Maintenance of Filled Discharge Pipe**

3.5.H Maintenance of Filled Discharge Pipe

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

I. Average Planar Linear Heat Generation Rate

During steady state power operation, the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) 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, -2.

If at any time during 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 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.H Maintenance of Filled Discharge Pipe

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.

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 shall be checked daily during reactor fuel operation at ≥ 25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

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 full withdrawn)

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

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

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 surveillance 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.

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

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 Specification 4.3.C.1 and 4.3.C.2.

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

Amendment Nos. ~~67~~,~~85~~,125

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting value for MAPLHGR is shown in Tables 3.5.I-1, -2. The analyses supporting these limiting values is presented in Reference 1.

3.5.J. Linear Heat Generation Rate (LHGR)

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

The LHGR shall be checked daily during reactor operation at $\geq$ 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 R factor would have to be less than 0.241 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimimum 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 MPCR 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 APRM Setpoints

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

TABLE 3.5.I-1

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Types: P8DRB284L, QUAD+ and 8DRB284L

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.3
5,000	11.8
10,000	12.0
15,000	12.0
20,000	11.8
25,000	11.2
30,000	10.8
35,000	10.0
40,000	9.4

Table 3.5.I-2

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Types: P8DRB265H

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	12.0
25,000	11.6
30,000	11.2
35,000	10.9
40,000	10.5
45,000	10.0

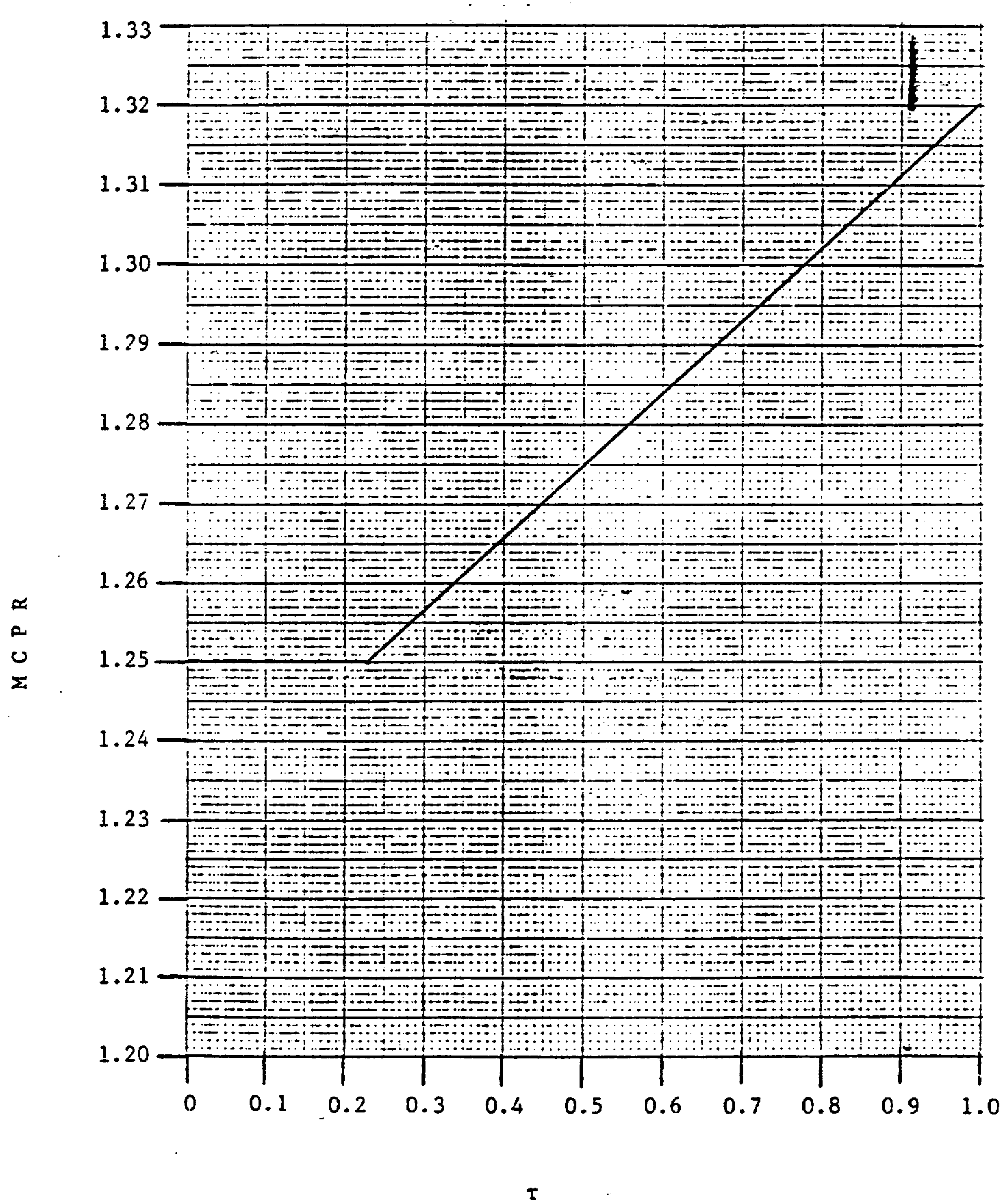

Figure 3.5.K-1

MCPR Limits for P8 x 8R/8 X 8R/ QUAD+

Amendment No. ~~88~~ ,125

3.6/4.6 BASES:

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 set points are within the ± 1 percent tolerance. The relief valves are tested in place once per operating cycle to establish that they will open and pass steam.

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.

REFERENCES

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)
2. Amendment 22 in response to AEC Question 4.2 of December 6, 1971.
3. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)
4. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilleland to F. E. Kruesi, August 29, 1973.
5. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A, and Addenda.

3.6.E/4.6.E Jet Pumps

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

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

Amendment No. 100, 125

TABLE 3.7.A
PRIMARY CONTAINMENT ISOLATION VALVES

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
1	Main steamline isolation valves (FCV-1-14, 26, 37, & 51; 1-15, 27, 38 & 52)	4	4	3 < T < 5	0	GC
1	Main steamline drain isolation valves (FCV-1-55 & 1-56)	1	1	15	O	GC
1*	Reactor Water sample line isolation valves	1	1	5	C	SC
2	RHRS shutdown cooling supply isolation valves (FCV-74-48 & 47)	1	1	40	C	SC
2	RHRS - LPCI to reactor (FCV-74-53 & 67)		2	30	C	SC
2	RHRS flush and drain vent to suppression chamber (FCV-74-102, 103, 119, & 120)		4	20	C	SC
2	Suppression Chamber Drain (FCV-75-57 & 58)		2	15	O**	GC
2	Drywell equipment drain discharge isolation valves (FCV-77-15A & 15B)		2	15	0	GC
2	Drywell floor drain discharge isolation valves (FCV-77-2A & 2B)		2	15	0	GC

**These valves are normally open when the pressure suppression head tank is aligned to serve the RHR and CS discharge piping and closed when the condensate head tank is used to serve the RHR and CS discharge piping. (See specification 3.5.H)

*These valves isolate only on reactor vessel low low water level (470") and main steam line high radiation of Group 1 isolations.

TABLE 3.7.B

TESTABLE PENETRATIONS WITH DOUBLE O-RING SEALS

Penetration No.	Identification
X-1A	Equipment Hatch
X-1B	Equipment Hatch
X-4	Head Access, Drywell
X-6	CRD Removal Hatch
X-25	Flange on 64-18
X-25	Flange on 64-19
X-25	Flange on 84-8A
X-25	Flange on 84-8D
X-26	Flange on 64-31
X-26	Flange on 64-34
X-35A	TIP Drive
X-35B	TIP Drive
X-35C	TIP Drive
X-35D	TIP Drive
X-35E	TIP Drive
X-35F	TIP Indexer Purge
X-35G	Spare
X-47	Power Operation Test
X-200A	Suppression Chamber Access Hatch
X-200B	Suppression Chamber Access Hatch
-	Drywell Head
-	Shear Lug No. 1
-	Shear Lug No. 2
-	Shear Lug No. 3
-	Shear Lug No. 4
-	Shear Lug No. 5
-	Shear Lug No. 6
-	Shear Lug No. 7
-	Shear Lug No. 8
X-205	Flange on 64-20
X-205	Flange on 64-21
X-205	Flange on 84-8B
X-205	Flange on 84-8C
X-205	Flange on 76-18
X-205	Flange on 76-19
X-223	Suppression Chamber Access Hatch
X-231	Flange on 64-29
X-231	Flange on 64-32

TABLE 3.7.C
TESTABLE PENETRATIONS WITH TESTABLE BELLOWS

X-7A	-	Primary Steamline	X-11	-	Steamline to HPCI Turbine
X-7B	-	Primary Steamline	X-12	-	RHR Shutdown Supply Line
X-7C	-	Primary Steamline	X-13A	-	RHR Return Line
X-7D	-	Primary Steamline	X-13B	-	RHR Return Line
X-8	-	Primary Steamline Drain	X-14	-	Reactor Water Cleanup Line
X-9A	-	Feedwater Line	X-16A	-	Core Spray Line
X-9B	-	Feedwater Line	X-16B	-	Core Spray Line
X-10	-	Steamline to RCIC Turbine	X-17	-	Blank

TABLE 3.7.E

PRIMARY CONTAINMENT ISOLATION VALVES WHICH TERMINATE
BELOW THE SUPPRESSION POOL WATER LEVEL

Valve	Valve Identification
12-738	Auxiliary Boiler to RCIC
12-741	Auxiliary Boiler to RCIC
43-28A	RHR Suppression Chamber Sample Lines
43-28B	RHR Suppression Chamber Sample Lines
43-29A	RHR Suppression Chamber Sample Lines
43-29B	RHR Suppression Chamber Sample Lines
71-14	RCIC Turbine Exhaust
71-[illegible]	RCIC Vacuum Pump Discharge
71-[illegible]	RCIC Turbine Exhaust
71-592	RCIC Vacuum Pump Discharge
73-23	HPCI Turbine Exhaust
73-24	HPCI Turbine Exhaust Drain
73-603	HPCI Turbine Exhaust
73-609	HPCI Exhaust Drain
74-722	RHR
75-57	Core Spray to Auxiliary Boiler
75-58	Core Spray to Auxiliary Boiler
	Core Spray to Auxiliary Boiler

Amendment No. ~~85~~,125

TABLE 3.7.F

PRIMARY CONTAINMENT ISOLATION VALVES LOCATED IN WATER SEALED SEISMIC CLASS 1 LINES

Valve	Valve Identification
74-53	RHR LPCI Discharge
74-54	RHR
74-57	RHR Suppression Chamber Spray
74-58	RHR Suppression Chamber Spray
74-60	RHR Drywell Spray
74-61	RHR Drywell Spray
74-67	RHR LPCI Discharge
74-68	RHR LPCI Discharge
74-71	RHR Suppression Chamber Spray
74-72	RHR Suppression Chamber Spray
74-74	RHR Drywell Spray
74-75	RHR Drywell Spray
75-25	Core Spray Discharge
75-26	Core Spray Discharge
75-53	Core Spray Discharge
75-54	Core Spray Discharge

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

Browns Ferry unit 2 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 4 QUAD+ demonstration assemblies, 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).

Amendment Nos. ~~33~~,~~46~~,~~58~~,~~85~~,125

B. Source Tests

Results of required leak tests performed on sources if the tests reveal the presence of 0.005 microcurie or more of removable contamination.

C. Special Reports (in writing to the Director of Regional Office of Inspection and Enforcement).

1. Reports on the following areas shall be submitted as noted:

a.	Secondary Containment Leak Rate Testing (5)	4.7.C	Within 90 days of completion of each test.
b.	Fatigue Usage Evaluation	6.6	Annual Operating Report
c.	Relief Valve Tailpipe Instrumentation	3.2.F	Within 30 days after inoperability of thermocouple and acoustic monitor on one valve.
d.	Seismic Instrumentation Inoperability	3.2.J.3	Within 10 days after 30 days of inoperability
e.	Meteorological Monitoring Instrumentation Inoperability	3.2.I.2	Within 10 days after 7 days of inoperability
f.	Primary Containment Integrated Leak Rate Testing	4.7.A.2	Within 90 days of completion of each test.
	High-Range Primary Containment Radiation Monitors	3.2.F	Within 7 days after 7 days of inoperability
	High-Range Gaseous Effluent Radiation Monitor	3.2.F	Within 7 days after 7 days of inoperability

D. Special Report (in writing to the Director of Regional Office of Inspection and Enforcement)

Data shall be retrieved from all seismic instruments actuated during a seismic event and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be submitted within 10 days after the event describing the magnitude, frequency spectrum, and resultant effect upon plant features important to safety.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 125 TO FACILITY OPERATING LICENSE NO. DPR-52

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-260

1.0 INTRODUCTION

By letter dated August 23, 1984 (TVA BFNP TS-199), as supplemented September 4 and November 13, 1984, April 3, May 8, June 27, November 20 and December 30, 1985, and April 29, 1986, the Tennessee Valley Authority (the licensee or TVA) requested an amendment to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit 2. The proposed amendment would change the Technical Specifications (TS) of the operating license to: (1) modify the core physics, thermal and hydraulic limits to be consistent with the reanalyses associated with replacing about one-third of the core during the Cycle 6 core reload outage and (2) reflect changes in various specifications as a result of plant modifications performed during the outage. In addition, TVA has updated the TS pages involved and made administrative corrections.

The areas involved in the amendment are as follows:

A. Core related changes

B. Changes related to torus modifications

C. Miscellaneous plant modifications

1. Reactor protection system (RPS) modification
2. Scram discharge instrument volume
3. Analog trip system
4. Scram permission pressure switches
5. Drywell temperature and pressure
6. TMI Action plan items (NUREG-0737)
7. Testable penetrations
8. Redundant air supply to the drywell
9. Demineralized water isolation valve
10. Residual heat removal (RHR) head spray

D. Administrative changes

2.0 EVALUATION

A. Core related changes

TVA made application to amend the Technical Specifications of Browns Ferry Nuclear Plant, Unit 2. The changes were required, in part, in order to permit the reloading and operation of Unit 2 for Cycle 6. In support of the application TVA submitted a Reload Licensing Report (Reference 1). The staff has reviewed this document and prepared the following evaluation of those aspects of the application pertaining to the reload.

Reload Description

For Cycle 6, 300 irradiated fuel assemblies will be removed from the core and replaced by 296 General Electric P8X8R assemblies and 4 Westinghouse designed QUAD + demonstration assemblies. In addition, the reload analysis has been performed by TVA, with the exception of the LOCA analysis which has been done by General Electric. The demonstration program has been described and analyses performed on the effect of the QUAD + assemblies on the core parameters by Westinghouse Nuclear Energy System, the manufacturer of the assemblies. TVA has submitted a report, WCAP-10507, "QUAD + Demonstration Assembly Report" (Reference 2) for the description of the program and its effects. The use of increased core flow is planned for Cycle 6. Analyses were performed for both 100 percent and 105 percent of rated flow and the most conservative results were used in determining the operating limits.

Fuel Mechanical Design

The P8X8R assemblies to be loaded into the core are identical to those inserted in Cycle 5. They are standard General Electric BWR fuel assemblies which are described in the GESTAR document (Reference 3) and we conclude that no further review of these assemblies is required. The mechanical design of the four QUAD + assemblies is described in Reference 2. That document also describes the fuel rod design analysis. The acceptability of these analyses for Lead Test Assemblies is the subject of a separate evaluation (Attached). That evaluation concludes that the QUAD + assemblies may use the various fuel rod design criteria of the P8X8R fuel on an interim basis for the Lead Test Assemblies.

Nuclear Design

This reload is the first one performed for Unit 2 by the licensee. The analysis methods used by TVA are described in References 4, 5 and 6. These reports have been reviewed and approved by the staff for use in such analyses. The results of the analyses are reported in Reference 1. The shutdown margin is calculated to be 1.0 percent reactivity change at the point in the cycle at which it is a minimum. This value exceeds the Technical Specification requirement of 0.38 percent and is acceptable. The standby Liquid Control System provides a shutdown margin of 1.8

percent reactivity change with a boron concentration of 600 ppm boron. This is an acceptable value. Reactivity coefficients are not used in the performance of transients by TVA. However, a void coefficient is obtained in the process of collapsing from 3-D to 1-D cross-sections. This value is in the range of those customarily obtained for BWR reload cores and is acceptable. The effect of the presence of the four Quad + assemblies on the neutronic behavior of the core is discussed in Reference 2, which is the subject of a separate evaluation (Attached). That evaluation concludes that the presence of the four QUAD + assemblies has a negligible effect on core neutronics. TVA has performed cycle specific analyses and concurs with the conclusions of the Westinghouse report. We conclude that the nuclear design and analysis of the Cycle 6 core are acceptable.

Thermal-Hydraulic Design

The thermal-hydraulic analysis of the Browns Ferry Unit 2 Cycle 6 reload has been reviewed to determine whether acceptable thermal-hydraulic limits have been met, whether acceptable analytical methods were used and whether the core exhibits thermal-hydraulic stability.

Safety Limit MCPR

The GEXL Critical Heat Flux Correlation is used to obtain the value of the safety limit MCPR. This correlation has been previously used for Browns Ferry Unit 2 and continues to be acceptable. The value of 1.07 for the safety limit MCPR is generic for BWR reloads and is acceptable.

Operating Limit MCPR

The procedures and techniques used to obtain the value of the operating limit MCPR are described in Reference 7 which has been reviewed and approved by the staff. The anticipated transients are analyzed to determine that which yields the largest reduction in CPR. That value is then added to the safety limit value (1.07) to obtain the operating limit MCPR. For the pressurization events both Option A and Option B limits are obtained. The results were calculated for the P8X8R fuel. The QUAD + fuel will be loaded into non-limiting core locations and monitored to the same operating MCPR limits.

Operation at 105 Percent of Rated Flow

The licensee proposes to operate at core flow rates up to 105 percent of rated flow for Cycle 6. Such operation has been approved for Cycle 5 in Browns Ferry Unit 2 and it continues to be acceptable for Cycle 6. Analysis of Cycle 6 operation has taken into account such operation.

Core Thermal-Hydraulic Stability

TVA uses a computerized model for analysis of boiling water reactor (BWR) stability for Cycle 6 of Browns Ferry Unit 2. The analysis model is based on the LAPUR computer code and is applicable to both core and channel hydrodynamic stability. It is the same model which was used for the analysis of the previously approved Browns Ferry Unit 3 Cycle 6 reload.

The model proposed by TVA has been under review by the staff. The safety evaluation of this model has not yet been issued but the review has progressed sufficiently for the staff to approve the TVA analysis of Cycle 6 of Browns Ferry Unit 2 for the following reasons.

1. The only significant change in fuel loading between Cycle 6 of Browns Ferry Unit 2 and the previously approved and currently operating Cycle 5 of Unit 2, is the addition of the four QUAD + demonstration assemblies. The stability characteristics of these assemblies were reviewed separately (see next section) and found acceptable.

2. The decay ratio as calculated by the TVA model for Cycle 6 of Browns Ferry Unit 2 is .71, which is lower than the calculated decay ratio (.73) of the previously approved Cycle 6 of Browns Ferry Unit 3.

3. The TVA model does a good job in predicting the results of the Peach Bottom Thermal-Hydraulic Stability Tests.

Presence of QUAD + Assemblies

The thermal-hydraulic performance of the QUAD + assemblies is discussed in Reference 2. The evaluation of that reference (Attached) concludes that use of QUAD + bundles as demonstration assemblies is acceptable provided that the guidelines of Section 4.1 of Reference 2 are followed and that a cycle specific analysis shows at least a margin of 20 percent in power between the QUAD + assembly and the lead assembly at full power and flow conditions. TVA has confirmed that the guidelines were followed and performed analyses to show that a 27 percent power margin exists for Cycle 6. The staff asked Westinghouse to show that the stability characteristics of the QUAD + assemblies are acceptable for inclusion in the Browns Ferry Unit 3 Cycle 6 core. The results of Westinghouse's analytical evaluation which qualifies the QUAD + stability margin is presented in Reference 2. The focus of this evaluation is on individual channel stability since the small number of QUAD + demonstration assemblies in the core will not have any significant impact on the core average parameters and hence not affect overall core stability. The Westinghouse analysis show the QUAD + assemblies to have an additional margin of 0.15 in decay ratio when compared to the P8X8R fuel already in the core. The Westinghouse evaluation used parametric analyses based on published data to quantify the relative stability margin of the QUAD + demonstration assembly compared to the P8X8R fuel and did not perform detailed stability calculations for the QUAD + assembly itself.

The staff reviewed the analysis performed by Westinghouse in Reference 2 and has found it to be a reasonable method for approximating the stability margin for the QUAD + assembly. While the staff finds that such an approach is acceptable for the limited number (4) of QUAD + assemblies in the core it is very approximate and considerably more detailed calculations would be required to justify a full reload of QUAD + assemblies. We conclude that the thermal-hydraulic design and analysis for Browns Ferry Unit 2 Cycle 6 are acceptable.

Transient and Accident Analysis

Core-wide pressurization transients were analyzed with the TVA-RETRAN (Reference 7) code which has been reviewed and approved by the staff. The two conditions cited in the review use of the COMETHE-III J code and approval of the parent RETRAN code, has been satisfied. Use of TVA-RETRAN is therefore acceptable.

The nonpressurization events were analyzed with the three dimensional core simulator code (Reference 5) since these are either steady state events or very slow transients. The limiting pressurization transient is the Load Rejection Without Bypass and the limiting nonpressurization events are the Loss of Feedwater heater and Mislocated Bundle Error. Since the replacement fuel is identical to some of the fuel already present in the core, reanalysis of the LOCA event was not required. Reference 2 presents analyses to show that the MAPLHGR limits for the P8DRB284L assemblies can be conservatively applied to the QUAD + assemblies. The rod drop accident analysis was performed with the methodology described in Reference 8. This methodology was approved for use in the Cycle 6 reload analysis for Browns Ferry Unit 3 and is acceptable for Unit 2. The result of the analysis for Cycle 6 of Browns Ferry Unit 2 is 152 calories per gram peak fuel enthalpy. This value meets our acceptance criterion of 280 calories per gram for this event and is acceptable.

Technical Specification Changes

Scram Permissive Pressure Switches at 1055 PSIG

Current Technical Specifications require the main steam line isolation valve closure and the turbine condenser low vacuum scram functions to be operable in the refuel, startup/standby, and run modes. However, these trips are bypassed in the refuel and startup/standby modes unless the reactor pressure is greater than 1055 psig. Since the core is protected by a high pressure trip at 1055 psig in all modes the two scram functions serve no useful purpose in the refuel and startup/hot standby modes. TVA proposes to delete the requirement for operability of the scram functions in those modes and to remove the bypass function. As a result of our review of this area of operation, we agree that these scram requirements accomplish no useful purpose in these modes. We conclude that the proposed Technical Specification change is acceptable.

MCPR-MAPLHGR Specifications

The operating limit MCPR as a function of average scram time, τ has been altered to account for the Cycle 6 reload. The proposed curve (Figure 3.5.K-1) is consistent with the value given in the reload report (Reference 1) and is acceptable.

The MAPLHGR tables have been revised by deleting those for fuel types no longer present in the core and consolidating the data into two tables, 3.5.I-1 and 3.5.I-2. No changes have been made in the MAPLHGR values. The values for the P8DRB284L type are to be used for the QUAD + fuel. Such use is justified in Reference 2 for demonstration assemblies and is acceptable.

Reference in Bases

At various locations, the Technical Specification Bases have been revised to reflect the fact that the safety analyses were performed by TVA. These revisions are acceptable.

Based on the review described above, we conclude that Browns Ferry Unit 2 may be loaded and operated for Cycle 6. This includes the presence of four QUAD + bundles as lead test assemblies. This conclusion is based on the following:

1. The safety analyses have been performed by previously approved methods and procedures, except for those directly relating to the demonstration assemblies.

2. The use of the demonstration assemblies has been approved (see Attached evaluation) subject to certain conditions. These conditions have been met for Browns Ferry 2 Cycle 6.

3. The Cycle 6 core meets all the staff's acceptance criteria.

B. <u>Changes Related to Torus Modifications</u>

One of the changes to the TS is to revise the tables that list the surveillance instrumentation associated with the suppression pool bulk temperature. This modification provides an improved torus temperature monitoring system which consists of 16 sensors. This will provide a more accurate indication of the torus water bulk temperature as required by NUREG-0661 and will replace the suppression chamber water temperature instruments presently listed in the TS. This change has been previously approved for Unit 3 by Amendment No. 78 dated August 27, 1984.

The change to the TS are necessary follow up actions essential to the implementation of this improvement. The changes to the TS place operability and calibration requirements on the new temperature monitoring system. Since these are new instruments, the surveillance requirements are not presently in the TS.

We have reviewed this proposed change and find it consistent with NRC guidance and it is, therefore, acceptable.

C. <u>Miscellaneous plant modifications</u>

1. Reactor Protection System (RPS) Modifications.

By letter dated August 7, 1978, the Commission advised TVA that during review of Hatch Unit 2, the staff had identified certain deficiencies in the design of the voltage regulator system of the motor generator sets which supply power to the reactor protection system (RPS). Pursuant to 10 CFR 50.54(g), TVA was required to evaluate the RPS power supply for Browns Ferry 1, 2 and 3 in light of the information set forth in our letter. By letter dated September 24, 1980, the staff informed TVA (and most other BWRs) that "we have determined that modifications should be performed to provide fully redundant Class IE protection at the interface of non-Class IE power supplies and RPS." The staff also advised TVA that "we have found that the conceptual design proposed by the General Electric Company and the installed modification on Hatch are acceptable solutions to our concern." By letter dated December 4, 1980, TVA committed to install the required modifications. By letters dated October 30, 1981 and July 28, 1982, NRC sent TVA model Technical Specifications for electric power monitoring of the RPS design and modifications.

By letter dated June 27, 1985, the staff approved the TVA proposed design modifications to the RPS power supply system. During the current outage of Unit 2, the RPS is being modified to provide a fully redundant Class IE protection at the interface of the non-Class IE power supplies and the RPS. This will ensure that failure of a non-Class IE reactor protection power supply will not cause adverse interaction to the Class IE reactor protection system.

The Technical Specifications are being revised similar to the model TS provided to TVA to reflect the limiting conditions for operation and surveillance requirements associated with the RPS modifications. Page 42 is being modified to add a description of these sections in the Bases.

Based on our Safety Evaluation dated June 27, 1985, and the TS submitted, we find the proposed amendment acceptable.

2. Scram discharge instrument volume

The scram discharge instrument volumes (SDIVs) were modified to address inadequacies identified by the partial rod insertion eventon Browns Ferry Unit No. 3 in June 1980[1]. The modifications of interest to this Safety Evaluation involve replacing the scram discharge tank's float devices

(1) Briefly, an undetected accumulation of water in the SDV reduced the available free volume for discharge of scram water which inhibited insertion of the control rods. The level detection system utilized float type instruments and an inspection of the instruments turned up several floats that had been damaged. It could only be concluded that the floats had been subjected to harmful hydrodynamic forces.

with new electronic level instruments. These instruments will initiate a scram on high level.

Tables 4.1.A and 4.1.B were revised to reflect changes to the required surveillance testing on the two electronic level switches. The acceptability of the changes to the surveillance testing will be addressed in Section C-3 of this SE.

Based on our review, we conclude that the proposed modifications to the Technical Specifications in the instrumentation and controls area are acceptable. The basis for our determination is that the modifications are consistent with the staff guidelines as stated in the BWR Scram Discharge Safety Evaluation Report, dated December 1, 1980. In addition, these proposed modifications have been previously approved for Browns Ferry Unit 1, Amendment No. 93.

3. Analog trip system

The analog transmitter trip system (ATTS) is a new design for portions of the system instrumentation of the Reactor Protective System (RPS) of Boiling Water Reactors. It was developed by the General Electric Company (GE) and is being supplied as original equipment in later built BWRs (e.g., BWR 6). GE developed the ATTS to offset operating disadvantages of the digital sensor switches of the original safety system instrumentation. The principal objective of the ATTS is to improve sensor intelligence and reliability while enhancing testing procedures.

The design was adapted to Browns Ferry Unit 2 to replace the existing mechanical switches that sense drywell and reactor pressures with analog loops and to modify the reactor water level indication loops to improve the reliability, accuracy and response time of the instrumentation. Change in design basis, protective function, redundancy, trip point, and logic would not be involved or modified as a result of the equipment changes.

Basically, the licensee is proposing to replace Barton, Barksdale, Static-O-Ring, and Yarway instruments with Rosemount analog pressure transmitters and Rosemount analog trip units. Along with the system enhancement offered by the new electronic instrumentation, the licensee proposed to extend the maximum calibration interval to "once an operating cycle." This was based on the high reliability of the analog instrumentation systems.

The various calibration intervals (not the same as functional test intervals) being used at the plant are:

1) Once every 7 days
2) Once every 3 months
3) Once every 6 months
4) Once every 18 months
5) Once each refueling outage

The channel calibration once per operating cycle is less conservative than the present requirement for calibrations of some systems once every 18 months.

It has come to our attention that the duration of an operating cycle may not be adequately defined. Mid-cycle shutdown may occur such that an operating cycle may be extended well beyond the 18-month period which has been previously considered to be the longest operating cycle. The operating cycle time is dependent on the reload fuel design, which can vary between 12 and 18 months.

The primary factor in setting the calibration intervals is the drift of the transmitters and trip units. The total loop accuracy and the total loop drift are added to obtain the trip setpoint. In many cases, the manufacturer's specifications only provide drift values for 6 to 12 month intervals. These drift values must now be extrapolated linearly to provide for 18 months or longer calibration intervals.

Based on the above information, we concluded that the Technical Specification changes extending the calibration frequencies to "once/operating cycle" are acceptable if these calibration frequencies/intervals are limited to 18 months maximum. This limitation of once/operating cycle not to exceed 18 months for calibration intervals applies to the analog pressure transmitters and analog alarm units only and not to the mechanical pressure switches and their associated alarm units.

By letter dated April 29, 1986, TVA submitted supplement 3 to the amendment request dated August 23, 1984, which made the change from once per operating cycle to a minimum frequency of once per 18 months. Based on that supplement and our review we conclude that the proposed modifications are acceptable.

4. Scram permissive pressure switches

This has been covered in Section A above.

5. Drywell temperature and pressure

The drywell temperature and pressure surveillance instrumentation is being upgraded this outage to provide qualified, more reliable instrumentation. The TS, Tables 3.2.F and 4.2.F, have been revised to reflect new instrument numbers for the new upgraded drywell temperature and pressure instrumentation. The surveillance requirements remain the same. We have reviewed the proposed changes and based on our review find them acceptable.

6. TMI Action plan items (NUREG-0737)

In November 1980, the staff issued NUREG-0737, "Clarification of TMI Action Plan Requirements," which included all TMI Action Plan items approved by the Commission for implementation at nuclear power reactors. NUREG-0737 identifies those items for which Technical Specifications are required. A number of items which require Technical Specifications were scheduled for implementation after December 31, 1981. The staff provided guidance on the scope of Technical Specifications for all of these items in Generic Letter 83-36. Generic Letter 83-36 was issued to all Boiling Water Reactor licensees on November 1, 1983. In this Generic Letter, the staff requested licensees to:

a. review their facility's Technical Specifications to determine if they were consistent with the guidance provided in the Generic Letter, and

b. submit an application for a license amendment where deviations or absence of Technical Specifications were found.

By letter dated August 23, 1984, as supplemented, TVA responsed to Generic Letter 83-36 by submitting Technical Specification change request for Browns Ferry Unit 2. This evaluation covers the following TMI Action Plan items:

Noble Gas Effluent Monitor (II.F.1.1)

The licensee has supplemented the existing normal range monitors to provide noble gas monitoring in accordance with TMI Action Plan Item II.F.1.1. The proposed Technical Specifications for Noble Gas Effluent Monitor are consistent with the guidelines provided in Generic Letter 83-36. Therefore, we conclude that the TSs for Item II.F.1.1 are acceptable.

Sampling and Analysis of Plant Effluents (II.F.1.2)

The guidance provided by Generic Letter 83-36 requested that an administrative program should be established, implemented and maintained to ensure the capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident. The licensee has proposed TSs that are included with the TSs for Surveillance Instrumentation. The proposed TSs for sampling and analysis of plant effluents meet the intent of our guidance. Therefore, the proposed TSs are acceptable.

Drywell High-Range Radiation Monitor (II.F.1.3)

The licensee has installed two drywell radiation monitors in Browns Ferry Unit 2 that are consistent with the guidance of TMI Action Plan Item II.F.1.3. Generic Letter 83-36 provided guidance for limiting conditions for operation and surveillance requirements for these monitors. The licensee proposed TSs that are consistent with the guidance provided in Generic Letter 83-36. Therefore, we conclude that the proposed TSs for Item II.F.1.3 are acceptable.

Drywell Pressure Monitor (II.F.1.4)

Browns Ferry Unit 2 has been provided with two wide range channels for monitoring drywell pressure following an accident. The licensee has proposed TSs that are consistent with the guidelines contained in Generic Letter 83-36. Therefore, we conclude that the proposed TSs for drywell pressure monitors are acceptable.

Suppression Pool Water Level Monitor (II.F.1.5)

The suppression pool water level monitors at Browns Ferry Unit 2 provides the capability required by TMI Action Plan Item II.F.1.5. The proposed TSs contain limiting conditions of operation and surveillance requirements that are consistent with the guidance contained in Generic Letter 83-36. Therefore, we conclude that the proposed TSs for suppression pool water level monitors are acceptable.

7. Testable Penetrations

Modifications are being made to the flange side of 14 containment isolation valves which cannot be isolated from primary containment to be tested. This modification will provide two gaskets with a pressure tap between the gaskets to allow the flange to be leak tested. Operability of the valve will not be affected by this modification. Fourteen new testable penetrations resulted and they were added to the table of testable penetrations with double o-ring seals (Table 3.7.B). New surveillance requirements are also being added. This change was previously approved for Unit 3 by Amendment No. 78 dated August 27, 1984.

Several editorial changes were also made to this table. They include revising the identification name on several penetrations, adding a penetration that was tested but was inadvertently left out of the table and removing penetration X-213A which no longer exists. These changes are purely administrative. Other minor corrections to this table were also made. Penetration X-35G was listed in this table for "T.I.P Drives" and is being revised to reflect that it is a "Spare." The drywell head is being added to this table. It was inadvertently not listed, but was included in the surveillance program. We have reviewed the proposed changes and find that the changes bring Table 3.7.B into conformance with 10 CFR 50 Appendix J for all testable penetrations with double o-ring, and are acceptable.

8. Redundant Air Supply to the Drywell

This proposed change was removed by supplement 2 to the amendment request dated December 30, 1985.

9. Demineralized Water Isolation Valve

The TSs are revised to delete primary containment isolation valve 2-1143 of the demineralized water system. This valve isolated the demineralized water line to the torus ring header. The line is no longer used, so the valve will be removed and the line capped. No safety-related functions will be adversely affected by disconnecting this line. This was previously approved for Unit 3 by Amendment No. 78 dated August 27, 1984.

We have reviewed this change and find that the TS change replacing the valve by a cap that will not leak is acceptable.

10. Residual Heat Removal (RHR) Head Spray

Two isolation valves on the residual heat removal head spray line were removed from Unit 2. The head spray line was removed and the penetration capped. The TS are being revised to remove these valves from the table of valves to be tested. The change deletes primary containment isolation valves 74-77 and 74-78 of the RHR system head spray from Tables 3.7.A and 3.7.F. The removal of the head spray line is part of the Intergranular Stress Corrosion Cracking Study being done on Browns Ferry. No safety related functions will be adversely affected by disconnecting this line.

We have reviewed this change and find it acceptable.

D. Administrative Changes

Several administrative changes are being made to the Technical Specifications. These include revising the Table of Contents to reflect the change discussed above, and miscellaneous editorial changes such as to delete obsolete references, change bases to reflect the changes to the Technical Specifications, correct page numbers, correct typographical errors, etc. The surveillance requirements for the personnel air lock is being changed to be consistent with the surveillance for Units 1 and 3. The proposed change includes deletion of the reference to safety valves in conjunction with relief valves. The safety valves with unpiped discharge have been removed and replaced with relief valves.

3.0 ENVIRONMENTAL CONSIDERATIONS

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachment:
Evaluation

Principal Contributors: W. Brooks, G. Schwenk, J. Mauk, C. Patel, and M. Grotenhuis

Dated: August 19, 1986

References

1. Browns Ferry Nuclear Plant Reload Licensing Report, Unit 2, Cycle 6; TVA-RLR-002, July, 1984, as supplemented.

2. L. T. Mayhue, "QUAD + Demonstration Assembly Report", WCAP-10507 (Proprietary), March, 1984.

3. GESTAR II, "General Electric Standard Application for Reactor Fuel", NEDO-24011-A-4, January, 1982.

4. B. L. Darnell, et. al, "Methods for the Lattice Physics Analysis of LWR's," TVA-TR78-02A, April, 1978.

5. S. L. Forkner, et. al, "Three Dimensional Core Simulator Methods", TVA-TR78-03A, January, 1979.

6. "Verification of TVA Steady State BWR Physics Methods", TVA-TR79-01A, January, 1979.

7. "BWR Transient Analysis Model Utilizing the RETRAN Program", TVA-TR81-01, December, 1981.

8. Browns Ferry Nuclear Plant Reload Licensing Report, Unit 3, Cycle 6; TVA-RLR-001, January 1984.

ATTACHMENT

EVALUATION RELATING TO TOPICAL REPORT WCAP-10507

QUAD + DEMONSTRATION ASSEMBLY REPORT

1.0 INTRODUCTION

Westinghouse Nuclear Energy Systems has prepared a report, WCAP-10507, "QUAD - Demonstration Assembly Report" and submitted it to the NRC staff for information. Since TVA has referenced this report in its application for the Cycle 6 reload of Browns Ferry Unit 2, the staff has performed a "mini-review" of the report to evaluate the impact of including four of the QUAD + assemblies in the core as Lead Test Assemblies (LTAs). All aspects of the assembly performance are evaluated except that of thermal-hydraulic stability. That aspect is the subject of a separate evaluation. The evaluation follows.

2.0 EVALUATION

The QUAD + assembly has been designed to be a reload bundle for BWR/3 through BWR/6 cores with either "C" or "D" lattice designs. It is intended to provide reduction in fuel cycle costs along with increased thermal margins. Care has been taken to make the QUAD + assembly compatible with currently used BWR bundles, particularly the P8x8R design. Details of the design of the QUAD + assembly are held to be proprietary information by Westinghouse.

The report also includes a set of constraints to be used when inserting QUAD + assemblies into a core as lead test assemblies (LTAs). These include:

1. The QUAD + demonstration assembly will not become a lead assembly during normal operation.

2. The QUAD + demonstration assembly will not become limiting under transient conditions.

3. One QUAD + demonstration assembly should be placed adjacent to a Local Power Range Monitor (LPRM) string.

4. QUAD + demonstration assemblies should be loaded quarter-core symmetric.

5. QUAD + demonstration assemblies will not be loaded less than one row away from the analytically determined potential dropped rod.

6. QUAD + assemblies should preferably not be loaded next to control rods which are inserted in the power range of operation during the first cycle.

2.1 Fuel Mechanical Design

The QUAD + assembly is designed to have the same length as the standard BWR assembly but has slightly larger lateral dimensions. The QUAD + channel design has improved creep resistance compared to the standard design which ensures that an adequate gap between assemblies is maintained throughout core residence time to permit unhampered control rod movement. The upper and lower end fittings of the QUAD + design interface with the core internals in the same manner as those of the standard design.

The QUAD + assembly contains more fuel rods than the standard assembly. Each rod is smaller in diameter than the standard rod and is surrounded by Zircalloy cladding which has been specially treated to improve corrosion resistance. Six-inch blankets of natural uranium are provided at the top and bottom of the fuel stack and gadolinia is used in selected rods to improve radial power distribution and to control assembly reactivity. Top and bottom structures are designed to be compatible with the core internals. Grid spacers have been designed for low flow resistance and improved thermal performance. Fuel rod integrity is assured by evaluation to design criteria which prevent excessive fuel temperatures, excessive internal rod gas pressures due to fission gas release, clad flattening, fatigue, corrosion above clad material removal limits, and excessive cladding stresses and strains during normal operation and anticipated transients. The Westinghouse PAD fuel performance code was used for the analyses. This code has been approved for use with PWR fuel and we find its use for QUAD + fuel acceptable for lead test assemblies. This conclusion is based on the fact that large margins will be maintained between safety limits and expected fuel duty for the LTAs. The design evaluations show that the QUAD + fuel meets all the design criteria with margin.

2.2 Nuclear Design

The nuclear design of the QUAD + assemblies is described in the report. The assemblies were designed to be as nearly the same as the P8x8R replacement fuel as feasible. The assembly design and comparison calculations were performed with the PHOENIX and POLCA codes. These codes have not been formally reviewed by the staff but information has been provided by Westinghouse to show that the PHOENIX assembly code gives results consistent with their standard design methods. The POLCA code is sufficiently similar to the Westinghouse PALADON code to permit the conclusion that the 3-D comparisons are acceptable, particularly since the QUAD + assembly are located in non-limiting positions.

Comparisons were made between the two assemblies for:

- assembly reactivity (K_∞ vs exposure)

- local peaking factor
- void coefficient
- moderator temperature coefficient
- Doppler coefficient
- cold rodded and unrodded reactivity
- rod worth as a function of void content
- delayed neutron fraction and prompt neutron lifetime.

These calculations demonstrated that the QUAD + assembly characteristies were similar of those of the P8x8R assembly it is designed to replace, or were conservative with respect to it. Three dimensional calculations were performed with a QUAD + assembly replacing a standard assembly to confirm that such replacement has no significant effect on core behavior. The QUAD + assembly has a slightly flatter end-of-cycle axial power distribution than the standard assembly due to a smaller void coefficient in the former. LPRM readings near the QUAD + assembly were within 1 to 3 percent of those for a standard assembly assembly - well within the LPRM uncertainty. We conclude that substitution of four QUAD + assemblies for four standard assemblies will have negligible effect on the neutronic behavior of the core.

2.3 Thermal-Hydraulic Analysis

Acceptability of the thermal-hydraulic design is based on hydraulic compatibility of the QUAD + design with the 8x8R standard design and on acceptable CPR performance. It is claimed that flow tests have shown that virtually identical pressure drops exist across the two bundle types at rated core flow and power conditions, but no data are presented. Outer bypass flows and in-channel flows are also the same for the assembly types. Hydraulic compatibility is thereby assured. The CPR performance of the QUAD + assembly is calculated with the AA-74 correlation developed by ASEA-ATOM for an 8x8 fuel assembly. This use is supported by the observation that the improved spacer grid design results in extra CPR margin for the QUAD + assembly. The use of the GEXL safety limit value of 1.07 for the QUAD + assembly (used with the AA-74 correlation) is supported by the fact that the convoluted uncertainties of the parameters used in the CPR evaluation are essentially the same for the two correlations. However, the form of the two correlations is different and the conclusion that a limit of 1.07 applies to both may not be valid. Finally the GEXL correlation will be used for the QUAD + demonstration assemblies when operating in the reactor.

The two correlations have been compared for a number of plant operating conditions and shown to give similar results.

In order to obtain additional margin to CPR limits the guidelines listed in Section 1 above are designed to provide a 10-20 percent margin in power between the QUAD + assemblies and the leading assembly under normal operating core conditions.

2.4 Transient and Accident Analyses

2.4.1 Core-Wide Transients

The consequences of core-wide transients depend upon core-wide neutronics parameters, which are not altered significantly by the presence of the four

QUAD + assemblies. Thus the core response is not altered but the transient response of the assemblies themselves must be considered. For slow transients, such as loss of feedwater heater, the change in CPR for the QUAD + assembly is essentially the same as that for the P8x8R assembly. The rapid transients, such as load rejection without bypass, result in larger MCPR changes for the QUAD + fuel relative to the standard fuel. For a typical such transient the change in CPR of a QUAD + bundle could be as great as 8 percent larger than that for the standard bundle. As indicated in Section 4 above a margin of 10 to 20 percent is provided by following the guidelines given in Section 1. In view of the increased change in CPR during transients and the uncertainties in the applicability of the GEXL correlation to the QUAD + assembly we conclude that the generic margin of 10 to 20 percent is not sufficient. We will therefore require cycle specific calculations to assure that a margin of at least 20 percent is present.

2.4.2 Dropped Rod

The QUAD + assemblies will be placed in the core in positions at least one row away from the rod shown by analysis to have the greatest worth in the startup regime where the consequences of the rod drop accident are significant. The QUAD + assembly will thus not be limiting for this event.

2.4.3 Rod Withdrawal Error

The rod worths at power are smaller for QUAD + assemblies than for standard ones. In addition the QUAD + assemblies will be loaded into non-limiting locations. The intent of the demonstration program is to have the QUAD + assemblies in non-rodded locations at power. For these reasons the presence of the QUAD + assemblies will not affect the rod withdrawal error analysis.

2.4.4 Fuel Misloading Event

The mislocation and misorientation of QUAD + assembly has been analyzed. Since it has been designed to have essentially the same reactivity as the corresponding P8x8R assembly the analysis for the latter assembly is applicable. The flatter enrichment distribution factor of the QUAD + assembly result in smaller changes in LHGR and CPR for misorientation events than with the corresponding P8x8R assembly.

2.4.5 Loss of Coolant Accident (LOCA)

The QUAD + assembly has several features which tend to mitigate the consequences of the loss of coolant event when compared to the equivalent P8x8R assembly. These include improved radiation heat transfer characteristics and a thinner channel which is more easily quenched. The lower plate design tends to delay the voiding of the assembly leading to an extended film boiling period. For the same fuel bundle power, the linear heat generation rate in the fuel is lower. These reactors tend to reduce the peak cladding temperature in a LOCA compared to the equivalent P8x8R assembly. Thus it may be concluded that the

LOCA analysis performed for a core loaded with standard assemblies will be applicable to QUAD + fuel and that MAPLHGR limits obtained for the equivalent P8x8R assembly may be conservatively applied to the QUAD + assembly.

3.0 CONCLUSIONS

Based on the review which is described above we conclude that WCAP-10507 presents sufficient information to support the use of up to four QUAD + bundles as demonstration assemblies in BWR/3 through BWR/6 cores provided that:

1. The guidelines presented in Section 4.1.2 of WCAP-10507 are adhered to, and

2. Cycle specific analyses are performed to show that a margin of at least 20 percent in power exists between the QUAD + assembly and the lead assembly when the core is operating at full power, full flow conditions.

Any more extensive loading of QUAD + assemblies into BWRs will be subject to review in considerably greater depth than is described in this evaluation.