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R. Diggs
NSIC
TERA
Chairman, ASLAB

Docket No. 50-259

September 15, 1981

Mr. Hugh G. Parris
Manager of Power
Tennessee Valley Authority
500 Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris;

The Commission has issued the enclosed Amendment No. 76 to Facility License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit No. 1. This amendment changes the Technical Specifications in response to your request of April 29, 1981 (TVA BFNP TS 161), as supplemented by your letters of June 12, 1981 and July 13, 1981. The changes to the Technical Specifications (1) incorporate the limiting conditions for operation of the facility in the fifth fuel cycle following the current refueling outage, (2) reflect new primary containment atmospheric monitoring instrumentation installed during this outage, and (3) reflect modifications which we required to be made to the torus.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED BY
/S/

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

RECEIVED
SEP 28 1981
U.S. NUCLEAR REGULATORY COMMISSION

Enclosures:
1. Amendment No. 76 to DPR-33
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

no legal objection form of amdt + notice reviewed; content of SER not reviewed

OFFICE	DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:OR	OELD	CPB	
SURNAME	SNorris	RJClark:ms	TAIppolito	TMNovak	LLS		
DATE	9/14/81	9/9/81	9/14/81	9/14/81	9/14/81	9/11/81	

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NRC

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USGPO: 1981—335

Mr. Hugh G. Parris

cc:

H. S. Sanger, Jr., Esquire
General Counsel
Tennessee Valley Authority
400 Commerce Avenue
E 11B 33C
Knoxville, Tennessee 37902

Mr. Ron Rogers
Tennessee Valley Authority
400 Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Mr. Charles R. Christopher
Chairman, Limestone County Commission
P. O. Box 188
Athens, Alabama 35611

Ira L. Myers, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36104

Mr. H. N. Culver
249A HBD
400 Commerce Avenue
Tennessee Valley Authority
Knoxville, Tennessee 37902

Athens Public Library
South and Forrest
Athens, Alabama 35611

Director, Office of Urban & Federal Affairs
108 Parkway Towers
404 James Robertson Way
Nashville, Tennessee 37219

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street
Atlanta, Georgia 30308

Mr. Robert F. Sullivan
U. S. Nuclear Regulatory Commission
P. O. Box 1863
Decatur, Alabama 35602

Mr. John F. Cox
Tennessee Valley Authority
W9-D 207C
400 Commerce Avenue
Knoxville, Tennessee 37902

Mr. Herbert Abercrombie
Tennessee Valley Authority
P. O. Box 2000
Decatur, Alabama 35602

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated April 29, 1981, as supplemented by letters dated June 12, 1981 and July 13, 1981 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 License No. DPR-33 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 76 , 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 of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 15, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 76

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

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

iii/iv	73/74	171/172
vii/viii	79/80	172a/172b
9/10	121/122	181/182
16	123/124	218/219
19/20	129/130	220/221
21/22	131/132	226/227
23/24	143/144	235a
25/26	145/146	249
29/30	157/158	250/251
31/32	159/160	252/253
47/48	169/170	260/261
		262/263
		266/267
		268
		269/270
		330/331

2. The underlined pages are those being changed; marginal lines on these pages indicate the revised area. The overleaf page is provided for convenience.

3. Add the following new pages:

160a
169a
251a
261a

Amendment No. 76

<u>LIST OF TABLES (Cont'd)</u>

Amendment No. 76

LIST OF ILLUSTRATIONS

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

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

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

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

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

(Note: These settings assume operation within the basic thermal hydraulic design criteria. These criteria are LHGR<13.4 kw/ft for 8x8, 8x8R, and P8x8R fuel, MCPR limits of Spec 3.5.k. If it is determined that either of these design criteria is being violated during operation, action shall be initiated within 15 minutes to restore operation within prescribed limits. Surveillance requirements for APRM scram setpoint are given in specification 4.1.B.

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

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

B. APRM Rod Block Trip Setting

The APRM Rod block trip setting shall be:

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
1.1 FUEL CLADDING INTEGRITY	2.1 FUEL CLADDING INTEGRITY
or core coolant flow is less than 10% of rated, the core thermal power shall not exceed 823 MWt (about 25% of rated thermal power).	$S_{RB} \leq (0.66W + 42\%)$ where: S_{RB} = Rod block setting is 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)
C. Whenever the reactor is in the shutdown condition reactor vessel, the water level shall not be less than 17.7 inches above the top of the normal active fuel zone.	C. Scram & isolation-reactor low water — $\geq$538 in. above vessel zero level D. Scram--turbine stop valve closure — $\leq$10 percent valve closure. E. Scram--turbine control valve 1. Fast Closure — Upon trip of the fast acting solenoid valves. 2. Loss of Control oil pressure — $\geq$ 550 psig F. Scram--low condenser vacuum — $\geq$ 23 inches Hg Vacuum G. Scram--main steam line isolation — $\leq$10 percent valve closure H. Main steam isolation valve closure--nuclear system low pressure — $\geq$825 psig

1.1 BASES

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

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to BFNP operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operating (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flow will always be greater than 4.56 psi. Analyses show that with a flow of $28X10^3$ lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than $28x10^3$ lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 Mwt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

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

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

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

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

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

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

In summary:

1. The licensed maximum power level is 3,293 MWt.
2. Analyses of transients employ adequately conservative values of the controlling reactor parameters.
3. The abnormal operational transients were analyzed to a power level of 3440 MWT.
4. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

The bases for individual set points are discussed below:

A. Neutron Flux Scram

1. APRM High Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during transients induced by disturbances, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses reported in Section 14 of the Final Safety Analysis Report demonstrated that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage. Therefore, use of a flow-biased scram provides even additional margin. Figure 2.1.2 shows the flow biased scram as a function of core flow.

An increase in the APRM scram setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM setting was selected because it provides adequate margin for the fuel cladding integrity safety limit yet allows operating margin that reduces the possibility of unnecessary scrams.

JAN 1 0 1979

2.1 BASES

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

2. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

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

3. IRM Flux Scram Trip Setting

The IRM System consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For

.1 BASES

3. IRM Flux Scram Trip Setting (Continued)

example, if the instrument were on range 1, the scram setting would be at 120 divisions for that range; likewise, if the instrument was on range 5, the scram setting would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. A scram at 120 divisions on the IRM instruments remains in effect as long as the reactor is in the startup mode. In addition, the APRM 15% scram prevents higher power operation without being in the RUN mode. The IRM scram provides protection for changes which occur both locally and over the entire core. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any safety limit is exceeded. For the case of a single control rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Quarter rod density is illustrated in paragraph 7.5.5 of the FSAR. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

B. APRM Control Rod Block

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recircuclation flow rate, and thus to protect against the condition of a MCPR less than 1.07. This rod block trip setting, which is automatically varried with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod withdrawal. The flow variable trip setting provides substantial margin

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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 100% 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 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 Scram

1. Fast Closure Scram

This turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection coincident with failures of the turbine bypass valves. The Reactor Protection System initiates a scram when fast closure of the control valves is initiated by the fast acting solenoid valves and in less than 30 milliseconds after the start of control valve fast closure. This is achieved by the action of the fast acting solenoid valves in 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. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first state pressure.

2.1 BASES

2. Scram on loss of control oil pressure

The turbine hydraulic control system operates using high pressure oil. There are several points in this oil system where a loss of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the generator load rejection scram, since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and high reactor pressure scrams. However, to provide the same margins as provided for the generator load rejection scram on fast closure of the turbine control valves, a scram has been added to the reactor protection system, which senses failure of control oil pressure to the turbine control system. This is an anticipatory scram and results in reactor shutdown before any significant increase in pressure or neutron flux occurs. The transient response is very similar to that resulting from the generator load rejection.

F. Main Condenser Low Vacuum Scram

To protect the main condenser against overpressure, a loss of condenser vacuum initiates automatic closure of the turbine stop valves and turbine bypass valves. To anticipate the transient and automatic scram resulting from the closure of the turbine stop valves, low condenser vacuum initiates a scram. The low vacuum scram set point is selected to initiate a scram before the closure of the turbine stop valves is initiated.

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

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

2.1 BASES

I. J. & K. Reactor low water level set point for initiation of HPCI and RCIC, closing main steam isolation valves, and starting LPCI and core spray pumps.

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

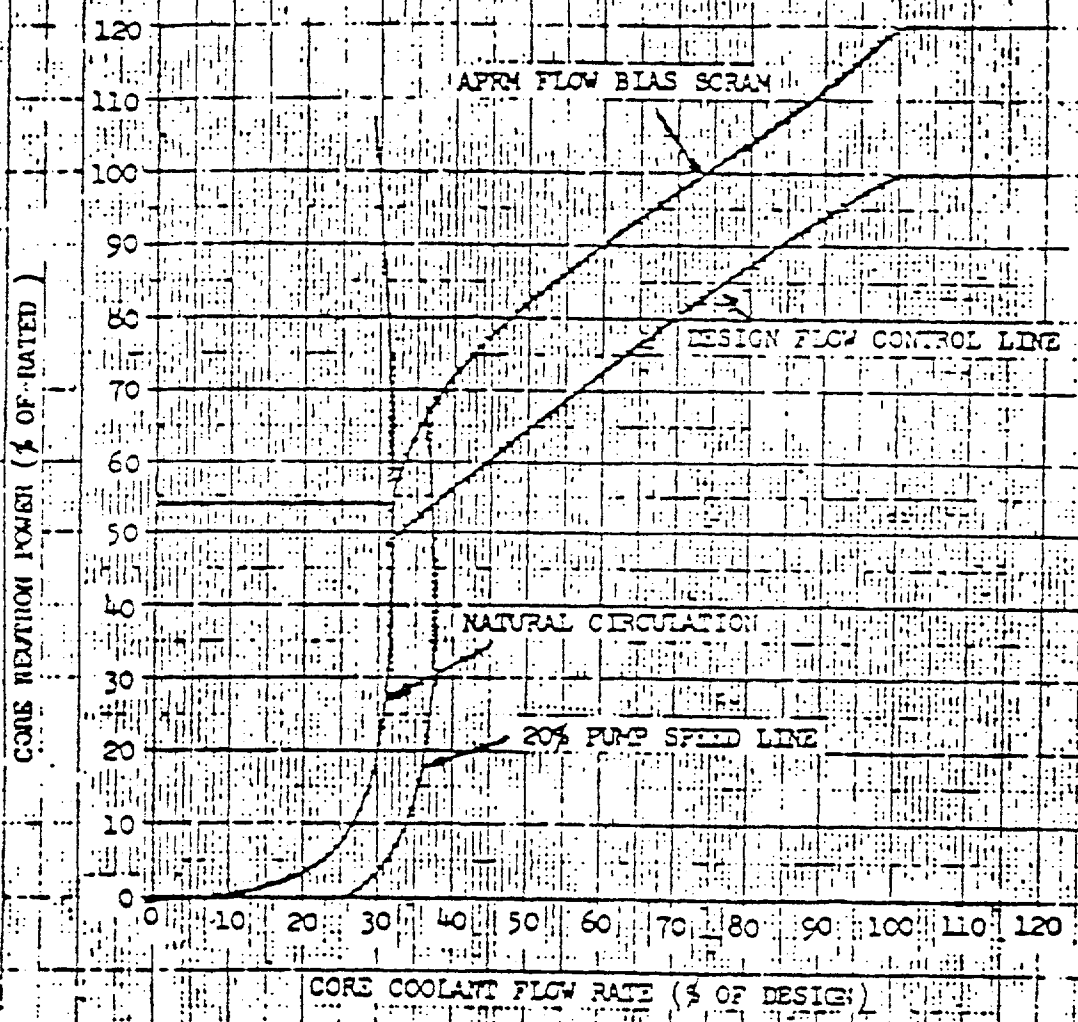
L. References

1. Linford, R. B. "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, Feb., 1973.

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

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

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



APRM FLOW BIAS SCRAM Vs. REACTOR CORE FLOW
FIG. 2.1-2

1.2 BASES

pressure monitor higher in the vessel. Therefore, following any transient that is severe enough to cause concern that this safety limit was violated, a calculation will be performed using all available information to determine if the safety limit was violated.

REFERENCES

1. Plant Safety Analysis (BFNP FSAR Section 14.0)
2. ASME Boiler and Pressure Vessel Code Section III
3. USAS Piping Code, Section B31.1
4. Reactor Vessel and Appurtenances Mechanical Design (BFNP FSAR Subsection 4.2)
5. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

To meet the safety design basis, thirteen relief valves have been installed on the unit with a total capacity of 83.9% of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steamline isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves operable, results in adequate margin to the code allowable overpressure limit of 1375 psig.

To meet the operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open)shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowable vessel over-pressure of 1375 psig.

3.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective

To assure the operability of the reactor protection system.

Specification

When there is fuel in the vessel, the setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.A.

4.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.A and 4.1.B respectively.

C. When it is determined that a channel is failed in the unsafe condition, the other RPS channe that monitor the same variable shall be functionally tested immediately before the trip sys tem containing the failure is tripped. The trip system containing the unsafe failure may untripped for short periods of time to allow functional testin of the other trip system. The trip system may be in the untripped position for no more than eight hours per functional test period for this testing.

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4.1 BASES

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

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup: i.e., the tests that are performed just prior to use of the instrument.

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

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

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

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month: i.e., in the period of a month a drift of 4% would occur and thus providing for adequate margin. For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

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

4.1 BASES

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

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

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

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TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum No. Operable Per Trip Sys (5)	Function	Trip Level Setting
2(1)	APRM Upscale (Flow Bias)	$\leq$ 0.66W + 42% (2)
2(1)	APRM Upscale (Startup Mode) (8)	$\leq$ 12%
2(1)	APRM Downscale (9)	$\geq$ 3%
2(1)	APRM Inoperative	(10_b)
1(7)	RBM Upscale (Flow Bias)	$\leq$ 0.66W + 40% (2)
1(7)	RBM Downscale (9)	$\geq$ 3%
1(7)	RBM Inoperative	(10_c)
3(1)	IRM Upscale (8)	$\leq$ 108/125 of full scale
3(1)	IRM Downscale (3)(8)	$\geq$ 5/125 of full scale
3(1)	IRM Detector not in Startup Position (8)	(11)
3(1)	IRM Inoperative (8)	(10_a)
2(1)(6)	SRM Upscale (8)	$\leq 1 \times 10^5$ counts/sec.
2(1)(6)	SRM Downscale (4)(8)	$\geq$ 3 counts/sec.
2(1)(6)	SRM Detector not in Startup Position (4)(8)	(11)
2(1)(6)	SRM Inoperative (8)	(10_a)
2(1)	Flow Bias Comparator	$\leq$ 10% difference in recirculation flows
2(1)	Flow Bias Upscale	$\leq$ 110% recirculation flow
1(1)	Rod Block Logic	N/A
2(1)	RSCS Restraint (PS-85-61A & PS-85-61B)	147 psig turbine first stage pressure (approximately 30% power)

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM, IRM, and APRM (Startup Mode), blocks need not be operable in "Run" mode, and the APRM (flow biased) and RBM rod blocks need not be operable in "Startup" mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter. If this condition lasts longer than seven days, the system with the inoperable channel shall be tripped. If the first column cannot be met for both trip systems, both trip systems shall be tripped.

2. W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (3293 MWt).

   See Specification 2.1 for APRM control rod block setpoint.

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is $\geq$100 cps and IRM above range 2.

5. One instrument channel, i.e., one APRM or IRM or RBM, per trip system may be bypassed except only one of four SRM may be bypassed.

6. IRM channels A, E, C, G, all in range 8 bypasses SRM channels A & C functions.

   IRM channels B, F, D, H, all in range 8 bypasses SRM channels B & D functions.

7. The following operational restraints apply to the RBM only.

   a. Both RBM channels are bypassed when reactor power is $\leq$ 30%.

   b. The RBM need not be operable in the "startup" position of the reactor mode selector switch.

   c. Two RBM channels are provided and only one of these may be bypassed from the console. An RBM channel may be out of service for testing and/or maintenance provided this condition does not last longer than 24 hours in any thirty day period.

   d. If minimum conditions for Table 3.2.C are not met, administrative controls shall be immediately imposed to prevent control rod withdrawal.

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TABLE 3.2.F

Surveillance Instrumentation

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

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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.

3.3.A REACTIVITY CONTROLS

c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

d. Control rods with a failed "Full-in" or "Full-out" position switch may be bypassed in the Rod Sequence Control System and considered operable if the actual rod position is known. These rods must be moved in sequence to their correct positions (full in on insertion or full out on withdrawal).

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5 x 5 array may be inoperable (at least 4 operable control rods must separate any 2 inoperable ones). If this Specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a shutdown condition within 24 hours.

B. Control Rods

1. Each control rod shall be coupled to its drive or completely inserted and the

4.3.A REACTIVITY CONTROLS

b. A second licensed operator shall verify the conformance to Specification 3.3.A.2.d before a rod may be bypassed in the Rod Sequence Control System.

c. When it is initially determined that a control rod is incapable of normal insertion an attempt to fully insert the control rod shall be made. If the control rod cannot be fully inserted, a shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined, highest worth control rod capable of withdrawal, full withdrawn, and all other control rods capable of insertion fully inserted.

d. The control rod accumulators shall be determined operable at least once per 7 days by verifying that the pressure and level detectors are not in the alarmed condition.

B. Control Rods

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

B Control Rods

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

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

3. a. Whenever the reactor is in the startup or run modes below 20% rated power the Rod Sequence Control System (RSCS) shall be operable except the RSCS constraints may be suspended by means of the individual rod bypass switches for

1 - special criticality tests, or

2 - control rod scram timing per 4.3.C.1.

When RSCS is bypassed on individual rods for these exceptions RWM must be operable per 3.3.B.3.c and a second licensed operator may not be used in lieu of RWM.

4.3.B Control Rods

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

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

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

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

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

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

3.3.B Control Rods

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

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

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

4.3.B Control Rods

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

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

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

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

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

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

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

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

3.3.B Control Rods

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

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

   a. Both RBM channels shall be operable: or

   b. Control rod withdrawal shall be blocked.

C. Scram Insertion Times

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

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

4.3.B Control Rods

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

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

C. Scram Insertion Times

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

3.3/4.3 BASES:

3. The Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to pre-specified sequences. All patterns associated with these sequences have the characteristic that, assuming the worst single deviation from the sequence, the drop of any control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in any pellet average enthalpy in excess of 280 calories per gram. An enthalpy of 280 calories per gram is well below the level at which rapid fuel dispersal could occur (i.e., 425 calories per gram). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Sections 3.6.6, 7.7.A, 7.16.5.3, and 14.6.2 of the FSAR and NEDO-10527 and supplements thereto.

In performing the function described above, the RWM and RSCS are not required to impose any restrictions at core power levels in excess of 20 percent of rated. Material in the cited reference shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 20 percent, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize individual control rod worth.

At power levels below 20 percent of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calorie per gram rod drop limit. In this range the RWM and the RSCS constrain the control rod sequences and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. Ref. Section 7.16.5.3 of the FSAR. They serve as a backup to procedure control of control rod sequences, which limit the maximum reactivity worth of control rods. Except during specified exceptions, when the Rod Worth Minimizer is out of service a second licensed operator can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural controls to assure conformance.

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

The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 20 percent, these devices force adherence to acceptable rod patterns. Above 20 percent of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 20 percent of rated power are imposed by power distribution requirements, as defined in Sections 3.5.I, 3.5.J, 4.5.I, and 4.5.J of these technical specifications. Power level for automatic bypass of the RSCS function is sensed by first stage turbine pressure.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It

does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

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

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

Scram Insertion Times

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

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

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

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

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

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

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

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

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

Objective

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

Specification

A. Core Spray System (CSS)

1. The CSS shall be operable:

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

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

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

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

Objective

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

Specification

A. Core Spray System (CSS)

1. Core Spray System Testing.

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

3.5.A Core Spray System (CSS)

2. If one CSS loop is inoperable, the reactor may remain in operation for a period not to exceed 7 days providing all active components in the other CSS loop and the RHR system (LPCI mode) and the diesel generators are operable.

3. If specification 3.5.A.1 or specification 3.5.A.2 cannot be met, the reactor shall be shutdown in the Cold Condition within 24 hours.

4. When the reactor vessel pressure is atmospheric and irradiated fuel is in the reactor vessel at least one core spray loop with one operable pump and associated diesel generator shall be operable, except with the reactor vessel head removed as specified in 3.5.A.5 or prior to reactor startup as specified in 3.5.A.1.

5. When irradiated fuel is in the reactor vessel and the reactor vessel head is removed, core spray is not required provided work is not in progress which has the potential to drain the vessel, provided the fuel pool gates are open and the fuel pool is maintained above the low level alarm point, and provided one RHRSW pump and associated valves supplying the standby coolant supply are operable.

4.5.A Core Spray System (CSS)

105 psi differential pressure between the reactor vessel and the primary containment.

e. Check Valve — Once/Operating Cycle

2. When it is determined that one core spray loop is inoperable, at a time when operability is required, the other core spray loop, the RHRS (LPCI mode), and the diesel generators shall be demonstrated to be operable immediately. The operable core spray loop shall be demonstrated to be operable daily thereafter.

LIMITING CONDITIONS FOR OPERATION

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

1. The RHRS shall be operable:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

1.

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

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

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

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

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3.5.F Reactor Core Isolation Cooling

2. If the RCICS is inoperable, the reactor may remain in operation for a period not to exceed 7 days if the HPCIS is operable during such time.

3. If specifications 3.5.F.1 or 3.5.F.2 are not met, an orderly shutdown shall be initiated and the reactor shall be depressurizeed to less than 122 psig within 24 hours.

G. Automatic Depressurization System (ADS)

1. Four of the six valves of the Automatic Depressurization System shall be operable:

(1) prior to a startup from a Cold Condition, or,

(2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 105 psig, except as specified in 3.5.G.2 and 3.5.G.3 below.

2. If three of the six ADS valves are known to be incapable of automatic operation, the reactor may remain in operation for a period not to exceed 7 days, provided the HPCI system is operable. (Note that the pressure relief function of these valves is assured by section 3.6.D of these specifications and that this specification only applies to the ADS function.) If more than three of the six ADS valves are known to be incapable of automatic operation, an immediate orderly shutdown shall be initiated, with the reactor in a hot shutdown condition in 6 hours and in a cold shutdown condition in the following 18 hours.

4.5.F Reactor Core Isolation Cooling

2. When it is determined that the RCICS is inoperable, the HPCIS shall be demonstrated to be operable immediately.

G. Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:

a. A simulated automatic actuation test shall be performed prior to startup after each refueling outage. Manual surveillance of the relief valves is covered, in 4.6.D.2.

2. When it is determined that more than two of the ADS valves are incapable of automatic operation, the HPCIS shall be demonstrated to be operable immediately and daily thereafter as long as Specification 3.5.G.2 applies.

3.5.G Automatic Depressurization System

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

H. Maintenance of Filled Discharge Pipe

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

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

P1-75-20	48 psig
P1-75-48	48 psig
P1-74-51	48 psig
P1-74-65	48 psig

4.5.G Automatic Depressurization System

H. Maintenance of Filled Discharge Pipe

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

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

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

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

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.I **Average Planar Linear Heat Generation Rate** During steady state power operation, the Maximum Average Planar Heat Generation Rate (MAPHGR) for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Tables 3.5.I-1 through 3.5.I-5. If at any time during steady state operation it is determined by normal. surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.	4.5.I **Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)** The MAPLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at ≥ 25% rated thermal power.
J. **Linear Heat Generation Rate (LHGR)** During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed 13.4 Kw/ft. If at any time during steady state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.	J. **Linear Heat Generation Rate (LHGR)** The LHGR for 8x8, 8x8R, and P8x8R fuel shall be checked daily during reactor operation at ≥ 25% rated thermal power.

3.5.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:

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

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

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

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

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

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

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

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

4.5.K. Minimum Critical Power Ratio (MCPR)

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

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

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

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

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

Limiting Conditions for Operation | Surveillance Requirements

3.5 Core and Containment Cooling Systems

L. APRM Setpoints

1. Whenever the core thermal power is $\geq$ 25% of rated, the ration of FRP/CMFLPD shall be $\geq$ 1.0, or the APRM scram and rod block setpoint equations listed in sections 2.1.A and 2.1.B shall be multiplied by FRP/CMFLPD as follows:

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

$$S_{RB} \leq (0.66 + 42\%) \left(\frac{FRP}{CMFLPD}\right)$$

2. When it is determined, that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.

3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to $\leq$ 25% of rated thermal power within 4 hours.

M. Reporting Requirements

If any of the limiting values identified in Specifications 3.5.I, J, K, or L.3 are exceeded and the specified remedial action is taken, the event shall be logged and reported in a 30-day written report.

4.5 Core and Containment Cooling Systems

L. APRM Setpoints

FRP/CMFLPD shall be determined daily when the reactor is $\geq$ 25% of rated thermal power.

3.5.J Linear Heat Generation Rate (LHGR)

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

The LHGR for 8x8, 8x8R, and P8x8R fuel shall be checked daily during reactor operation at ≥25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the, MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirments 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

The fuel cladding integrity safety limits of section 2.1 were based on a total peaking factor within design limits (FRP/CMFLPD ≥1.0). The APRM instruments must be adjusted to ensure that the core thermal limits are not exceeded in a degraded situation when entry conditions are less conservative than design assumptions.

3.5.M Reporting Requirements

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

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgement and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system; i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested each month to assure their operability. A simulated automatic actuation test once each cycle combined with monthly tests of the pumps and injection valves is deemed to be adequate testing of these systems.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, cause the outage, then the demonstration of operability should be thorough enough to assure that a generic problem does not exist. For example, if an out-of-service period was caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

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

Redundant operable components are subjected to increased testing during equipment out-of-service times. This adds further conservatism and increases assurance that adequate cooling is available should the need arise.

Maximum Average Planar LHGR, LHGR, and MCPR

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

3.5.N References

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

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

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

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

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

Table 3.5.I-1

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DB274L

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

Table 3.5.I-2

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DB274H

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

Table 3.5.I-3

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DRB265H

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

Table 3.5.I-4

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DRB265L and P8DRB265L

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

Table 3.5.I-5

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

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

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

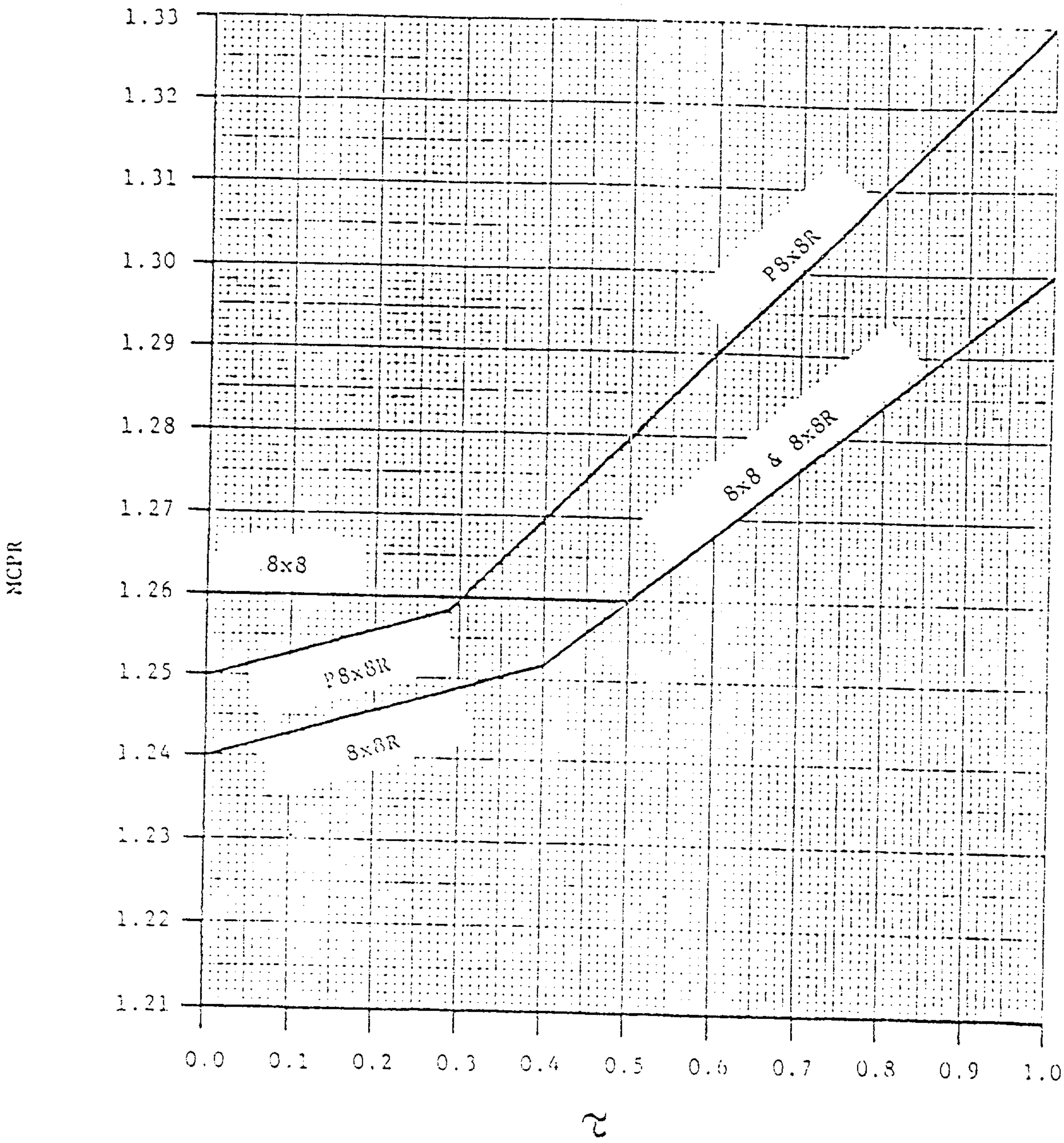

Figure 3.5.K-1
MCPR LIMITS*

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

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3.6.C Coolant Leakage

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

D. Relief Valves

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

E. Jet Pumps

1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be shutdown in the Cold Condition within 24 hours.

4.6.C Coolant Leakage

D. Relief Valves

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

2. Once during each operating cycle, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

3. The integrity of the relief/safety valve bellows shall be continuously monitored.

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

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the startup or run modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

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

3.6.E Jet Pumps

3.6.F Recirculation Pump Operation

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

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

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

G. Structural Integrity

1. The structural integrity of the primary system shall be

4.6.E Jet Pumps

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

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

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

4.6.F Recirculation Pump Operation

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

G. Structural Integrity

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

3.6/4.6 BASES:

The basis for the equilibrium coolant iodine activity limit is a computed dose to the thyroid of 36 rem at the exclusion distance during the 2-hour period following a steam line break. This dose is computed with the conservative assumption of a release of 140,000 lbs of coolant prior to closure of the isolation valves, and a X/Q value of 3.4 x 10^{-4} Sec/m^3.

The maximum activity limit during a short term transient is established from consideration of a maximum iodine inhalation dose less than 300 rem. The probability of a steam line break accident coincident with an iodine concentration transient is significantly lower than that of the accident alone, since operation of the reactor with iodine levels above the equilibrium value is limited to 5 percent of total operation.

The sampling frequencies are established in order to detect the occurrence of an iodine transient which may exceed the equilibrium concentration limit, and to assure that the maximum coolant iodine concentrations are not exceeded. Additional sampling is required following power changes and off-gas transients, since present data indicate that the iodine peaking phenomenon is related to these events.

3.6.C/4.6.C Coolant Leakage

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite a-c power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the unit should be shut down to allow further investigation and corrective action.

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

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

REFERENCES

1. Nuclear System Leakage Rate Limits (BFNP FSAR Subsection 4.10)

3.6.D/4.6.D Relief Valves

To meet the safety basis thirteen relief valves have been installed on the unit with a total capacity of 83.9% of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves operable, results in adequate margin to the code allowable overpressure limit of 1375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1375 psig.

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.

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

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

3.6.F/4.6.F Recirculation Pump Operation

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

These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Ten percent or ten snubbers whichever is less, represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. Those snubbers designated in Table 3.6.H as being in high radiation areas or especially difficult to remove need not be selected for functional tests provided operability was previously verified.

Snubbers of rated capacity greater than 50,000 lb. are exempt from the functional testing requirements because of the impracticability of testing such large units.

REFERENCES

1. Report, H. R. Erickson, Bergen Paterson to K. R. Goller, NRC, October 7, 1974, Subject: Hydraulic Shock Sway Arrestors

3.7 CONTAINMENT SYSTEMS

Applicability

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

Objective

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

Specification

A. Primary Containment

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

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

   -7.25" (0 psid differential pressure control)

   b. Maximum water level = -1"

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary and secondary containment integrity.

Objective

To verify the integrity of the primary and secondary containment.

Specification

A. Primary Containment

1. Pressure Suppression Chamber

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

3.7 CONTAINMENT SYSTEMS

6. Drywell-Suppression Chamber Differential Pressure

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

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

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

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

4.7 CONTAINMENT SYSTEMS

6. Drywell-Suppression Chamber Differential Pressure

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

3.7 CONTAINMENT SYSTEMS

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

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

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

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

4.7 CONTAINMENT SYSTEMS

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

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

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

TABLE 3.7.A
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	C	SC
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	Reactor vessel head spray isolation valves FCV-74-77, 78	1	1	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-74-57, 58		2	15	C	SC
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

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
3	Reactor water cleanup system supply isolation valves FCV-69-1, & 2	1	1	30	0	GC
3	Reactor water cleanup system return isolation valves FCV-69-12		1	60	0	GC
4	FCV 73-81 (Bypass around FCV 73-3)		1	10	0	GC
4	HPCIS steamline isolation valves FCV-73-2 & 3	1	1	20	0	GC
5	RCICS steamline isolation valves FCV-71-2 & 3	1	1	15	0	GC
6	Drywell nitrogen purge inlet isolation valves (FCV-76-18)		1	5	C	SC
6	Suppression chamber nitrogen purge inlet isolation valves (FCV-76-19)		1	5	C	SC
6	Drywell Main Exhaust isolation valves (FCV-64-29 and 30)		2	2.5	C	SC
6	Suppression chamber main exhaust isolation valves (FCV-64-32 and 33)		2	2.5	C	SC
6	Drywell/Suppression Chamber purge inlet (FCV-64-17)		1	2.5	C	SC
6	Drywell Atmosphere purge inlet (FCV-64-18)		1	2.5	C	SC

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TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (Sec.)	Normal Position	Action On Initiating Signal
6	Torus Hydrogen Sample Line Valves Analyzer A (FSV-76-55, 56)	1	1	NA	Note 1	SC
6	Torus Oxygen Sample Line Valves Analyzer A (FSV-76-53, 54)	1	1	NA	Note 1	SC
6	Drywell Hydrogen Sample Line Valves Analyzer A (FSV-76-49, 50)	1	1	NA	Note 1	SC
6	Drywell Oxygen Sample Line Valves Analyzer A (FSV-76-51, 52)	1	1	NA	Note 1	SC
6	Sample Return Valves - Analyzer A (FSV-76-57, 58)	1	1	NA	0	SC
6	Torus Hydrogen Sample Line Valves Analyzer B (FSV-76-65, 66)	1	1	NA	Note 1	SC
6	Torus Oxygen Sample Line Valves-Analyzer B (FSV-76-63, 64)	1	1	NA	Note 1	SC
6	Drywell Hydrogen Sample Line Valves-Analyzer B (FSV-76-59, 60)	1	1	NA	Note 1	SC
6	Drywell Oxygen Sample Line Valves-Analyzer B (FSV-76-61, 62)	1	1	NA	Note 1	SC
6	Sample Return Valves-Analyzer B (FSV-76-67, 68)	1	1	NA	0	SC

Note 1: Analyzers are such that one is sampling drywell hydrogen and oxygen (valves from drywell open- valves from torus closed) while the other is sampling torus hydrogen and oxygen (valves from torus open - valves from drywell closed)

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
6	Suppression Chamber purge inlet (FCV-64-19)		1	2.5	C	SC
6	Drywell/Suppression Chamber nitrogen purge inlet (FCV-76-17)		1	5	C	SC
6	Drywell Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-31)		1	5	C	SC
6	Suppression Chamber Exhaust Valve Bypass to Standby Gas Treatment System (FCV-64-34)		1	5	C	SC
6	Drywell/Suppression Chamber Nitrogen Purge Inlet (FCV-76-24)		1	5	C	SC
7	RCIC Steamline Drain (FCV-71-6A, 6B)		2	5	0	GC
7	RCIC Condensate Pump Drain (FCV-71-7A, 7B)		2	5	0	GC
7	HPCI Hotwell pump discharge isolation valves (FCV-73-17A, 17B)		2	5	C	SC
7	HPCI steamline drain (FCV-75-57, 58)		2	5	0	GC
8	TIP Guide Tubes (5)		1 per guide tube	NA	C	GC

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
	Standby liquid control system check valves CV 63-526 & 525	1	1	NA	C	Process
	Feedwater check valves CV-3-558, 572, 554, & 568	2	2	NA	0	Process
	Control rod hydraulic return check valves CV-85-576 & 573	1	1	NA	0	Process
	RHRS - LPCI to reactor check valves CV-74-54 & 68	2		NA	C	Process

TABLE 3.7.D (Continued)

Valves	Valve Identification	Test Medium	Test Method
69-1	RWCU Supply	Water (2)	Applied between 69-1, 69-500 and 10-505
69-2	RWCU Supply	Water (2)	Applied between 69-2, 69-500 and 10-505
71-2	RCIC Steam Supply	Air (1)	Applied between 71-2 and 71-3
73-81	HPCI Steam Supply Bypass	Air (1)	Applied between 73-2 and 73-3
71-3	RCIC Steam Supply	Air (1)	Applied between 71-2 and 71-3
71-39	RCIC Pump Discharge	Water (2)	Applied between 3-66, 3-568, 69-579, 71-39, and 85-576
73-2	HPCI Steam Supply	Air (1)	Applied between 73-2 and 73-3
73-3	HPCI Steam Supply	Air (1)	Applied between 73-2 and 73-3
73-44	HPCI Pump Discharge	Water (2)	Applied between 3-67, 3-554, and 73-44
74-47	RHR Shutdown Suction	Water (2)	Applied between 74-47, 74-754, 74-49, and 74-661
74-48	RHR Shutdown Suction	Water (2)	Applied between 74-48, 74-661 and 74-49
74-53	RHR LPCI Discharge	Water (2)	Applied between 74-53 and 74-55
74-57	RHR Suppression Chamber Spray	Water (2)	Applied between 74-57, 74-53, and 74-59
74-58	RHR Suppression Chamber Spray	Water (2)	Applied between 74-57, 74-58, and 74-59
74-60	RHR Drywell Spray	Water (2)	Applied between 74-60, 74-61
74-61	RHR Drywell Spray	Water (2)	Applied between 74-60, 74-61
74-67	RHR LPCI Discharge	Water (2)	Applied between 74-67 and 74-69
74-71	RHR Suppression Chamber Spray	Water (2)	Applied between 74-71, 74-72, and 74-73
74-72	RHR Suppression Chamber Spray	Water (2)	Applied between 74-71, 74-72, and 74-73
74-74	RHR Drywell Spray	Water (2)	Applied between 74-74, 74-75

TABLE 3.7.D (Continued)

'alves	Valve Identification	Test Medium	Test Method
74-75	RHR Drywell Spray	Water(2)	Applied between 74-74 and 74-75
74-77	RHR Head Spray	Water(2)	Applied between 74-77 and 74-78
74-78	RHR Head Spray	Water(2)	Applied between 74-77 and 74-78
74-661/662	RHR Shutdown Suction	Water(2)	Applied between 74-660 and 74-661/662
75-25	Core Spray Discharge	Water(2)	Applied between 75-25 and 75-27
75-53	Core Spray Discharge	Water(2)	Applied between 75-53 and 75-55
75-57	Core Spray to Auxiliary Boilers	Water(2)	Applied between 75-57 and 75-58
75-58	Core Spray To Auxiliary Boilers	Water(2)	Applied between 75-57 and 75-58
17	Drywell/Suppression Chamber Nitrogen Purge Inlet	Nitrogen(1)	Applied between 76-17, 76-18, 76-19
76-18	Drywell Nitrogen Purge Inlet	Nitrogen(1)	Applied between 76-17, 76-18, 76-19
76-19	Suppression Chamber Purge Inlet	Nitrogen(1)	Applied between 76-17, 76-18, 76-19
76-24	Drywell/Suppression Chamber Nitrogen Purge Inlet	Air(1)	Applied between 64-17, 64-18, 64-19, and 76-24
77-2A	Drywell Floor Drain Sump	Water(2)	Applied between 77-2A and 77-2B
77-2B	Drywell Floor Drain Sump	Water(2)	Applied between 77-2A and 77-2B
77-15A	Drywell Equipment Drain Sump	Water(2)	Applied between 77-15A and 77-15B
77-15B	Drywell Equipment Drain Sump	Water(2)	Applied between 77-15A and 77-15B
90-254A	Radiation Monitor Suction	Air(1)	Applied between 90-254A, 90-254B, and and 90-255
90-254B	Radiation Monitor Suction	Air(2)	Applied between 90-254A, 90-254B, and 90-255
-255	Radiation Monitor Suction	Air(2)	Applied between 90-254A, 90-254B, and 90-255

TABLE 3.7.D (Continued)

Valves	Valve Identification	Test Medium	Test Method
76-49	Containment Inerting	Air	Applied between inboard block valve and 76-49.
76-50	Containment Inerting	Air	Applied between inboard block valve and 76-50.
76-51	Containment Inerting	Air	Applied between inboard block valve and 76-51.
76-52	Containment Inerting	Air	Applied between inboard block valve and 76-52.
76-53	Containment Inerting	Air	Applied between inboard block valve and 76-53.
76-54	Containment Inerting	Air	Applied between inboard block valve and 76-54.
76-55	Containment Inerting	Air	Applied between inboard block valve and 76-55.
76-56	Containment Inerting	Air	Applied between inboard block valve and 76-56.
76-57	Containment Inerting	Air	Applied between inboard block valve and 76-57.
76-58	Containment Inerting	Air	Applied between inboard block valve and 76-58.
76-59	Containment Inerting	Air	Applied between inboard block valve and 76-59.
76-60	Containment Inerting	Air	Applied between inboard block valve and 76-60.
76-61	Containment Inerting	Air	Applied between inboard block valve and 76-61.
76-62	Containment Inerting	Air	Applied between inboard block valve and 76-62.
76-63	Containment Inerting	Air	Applied between inboard block valve and 76-63.
76-64	Containment Inerting	Air	Applied between inboard block valve and 76-64.
76-65	Containment Inerting	Air	Applied between inboard block valve and 76-65.
76-66	Containment Inerting	Air	Applied between inboard block valve and 76-66.
76-67	Containment Inerting	Air	Applied between inboard block valve and 76-67.
76-68	Containment Inerting	Air	Applied between inboard block valve and 76-68.

TABLE 3.7.D (Continued)

Valves	Valve Identification	Test Medium	Test Method
90-257A	Radiation Monitor Discharge	Air(1)	Applied between 90-237A and 90-257B
90-257B	Radiation Monitor Discharge	Air(1)	Applied between 90-237A and 90-257B
84-8A	Containment Atmospheric Dilution	Air	Applied between 84-8A and 84-600
84-8B	Containment Atmospheric Dilution	Air	Applied between 84-8B and 84-601
84-8C	Containment Atmospheric Dilution	Air	Applied between 84-8C and 84-603
84-8D	Containment Atmospheric Dilution	Air	Applied between 84-8D and 84-602
84-19	Containment Atmospheric Dilution	Air	Applied between 64-32, 64-33, 64-29, 64-30, and 84-19

(1) Air/nitrogen test to be displacement flow.
(2) Water test to be injection loss or downstream collection.

Valves	Valve Identification	Test Medium	Test Method
84-20	Main Exhaust to Standby Gas Treatment	Air(1)	Applied between 84-20, 64-141, 64-140, and 64-31
84-600	Main Exhaust to Standby Gas Treatment	Nitrogen(1)	Applied between 84-8A and 84-600
84-601	Main Exhaust to Standby Gas Treatment	Nitrogen	Applied between 84-8B and 84-601
84-602	Main Exhaust to Standby Gas Treatment	Nitrogen	Applied between 84-8C and 84-603
84-603	Main Exhaust to Standby Gas Treatment	Nitrogen	Applied between 84-8D and 84-602
64-141	Drywell Pressurization, Comp. Bypass	Air (1)	Applied between 64-141, 64-140, 64-30, and 84-20
64-140	Drywell Pressurization, Comp. Disc.	Air(1)	Applied between 64-141, 64-140, 64-31, and 84-20
64-139	Drywell Pressurization, Comp. Suction	Air(1)	Applied between 64-139, 64-141, and 64-34

1) Air/nitrogen test to be displacement flow
(2) Water test to be injection loss or downstream collection.

TABLE 3.7.E
SUPPRESSION CHAMBER INFLUENT LINES
STOP-CHECK GLOBE ISOLATION VALVES

Valves	Valve Identification	Test Medium	Test Method
71-14	RCIC Turbine Exhaust	Water	Apply between 71-14 and 71-580
71-32	RCIC Vacuum pump Discharge	Water	Apply between 71-32 and 71-592
73-23	HPCI Turbine Exhaust	Water	Apply between 73-23 and 73-603
73-24	HPCI Turbine Exhaust Drain	Water	Apply between 73-24 and 73-609

TABLE 3.7.F
CHECK VALVES ON SUPPRESSION CHAMBER INFLUENT LINES

Valves	Valve Identification	Test Medium	Test Method
71-580	RCIC Turbine Exhaust	Water	Apply between 71-14 and 71-580
71-592	RCIC Vacuum Pump Discharge	Water	Apply between 71-32 and 71-592
73-603	HPCI Turbine Exhaust	Water	Apply between 73-23 and 73-603
73-609	HPCI Exhaust Drain	Water	Apply between 73-24 and 73-609

TABLE 3.7.H (Continued)

X-107B	Spare (testable)
X-108A	Power
X-108B	CRD Rod Position Indic.
X-109	" " " "
X-110A	Power
X-110B	CRD Rod Position Indic.
X-230	Containment Air Monitoring System

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination, limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep offsite doses well below 10 CFR 100 limits.

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

Using the minimum or maximum water levels given in the specifications, containment pressure during the design basis accident is approximately 49 psig, which is below the maximum of 62 psig. The maximum water level indication of -1 inch corresponds to a downcomer submergence of 3 feet 7 inches and a water volume of 127,800 cubic feet with or 128,700 ft^3 without the drywell-suppression chamber differential pressure control. The minimum water level indication of -6.25 inches with differential pressure control and -7.25 inches without differential pressure control corresponds to a downcomer submergence of approximately 3 feet and a water volume of approximately 123,000 cubic feet. Maintaining the water level between these levels will assure that the torus water volume and downcomer submergence are within the aforementioned limits during normal plant operation. Alarms, adjusted for instrument error, will notify the operator when the limits of the torus water level are approached. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

BASES

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

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 200°F local.

Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressuirzed in a timely manner to avoid the regime of potentially high suppression chamber loadings.

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

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

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

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

Inerting

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

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

To ensure that the hydrogen concentration is maintained less than 4% following an accident, liquid nitrogen is maintained on-site for containment atmosphere dilution. About 2260 gallons would be sufficient as a 7-day supply, and replenishment facilities can deliver liquid nitrogen to the site within one day; therefore, a requirement of 2500 gallons is conservative. Following a loss of coolant accident the Containment Air Monitoring (CAM) System continuously monitors the hydrogen concentration of the containment volume. Two independent systems (a system consists of one hydrogen sensing circuit) are installed in the drywell and the torus. Each sensor and associated circuit is periodically checked by a calibration gas to verify operation. Failure of one system does not reduce the ability to monitor system atmosphere as a second independent and redundant system will still be operable.

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

5.0 MAJOR DESIGN FEATURES

5.1 SITE FEATURES

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

5.2 REACTOR

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

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

5.3 REACTOR VESSEL

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

5.4 CONTAINMENT

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

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

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

5.5 FUEL STORAGE

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

3. The k_{eff} of the spent fuel storage pool shall be less than or equal to 0.95.

C. Loads greater than 1000 pounds shall not be carried over spent fuel assemblies stored in the spent fuel pool.

5.6 SEISMIC DESIGN

The station class I structures and systems have been designed to withstand a design basis earthquake with ground acceleration of 0.2g. The operational basis earthquake used in the plant design assumed a ground acceleration of 0.1g (see Section 2.5 of the FSAR).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.76 TO FACILITY LICENSE NO. DPR-33

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-259

1.0 Introduction

By letter dated April 29, 1981 (TVA BFNP TS 161), which was supplemented by letters dated June 12, 1981 and July 13, 1981, the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit No. 1. The proposed amendment and revised Technical Specifications would (1) incorporate the limiting conditions for operation of the facility in the fifth fuel cycle following the fourth refueling of the reactor and (2) reflect new primary containment atmospheric hydrogen monitoring instrumentation being installed during the current refueling outage. In support of this reload application, TVA submitted a supplemental reload licensing document[1] prepared by the General Electric Company (GE), errata and addenda sheets to the Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 1[2] (originally issued September 1977) also prepared by GE and proposed changes to the Technical Specifications.

2.0 Discussion

Browns Ferry Unit No. 1 (BF-1) shutdown for its fourth refueling on April 11, 1981. BF-1 was initially fueled with 764 of the General Electric Co. (GE) 7 x 7 fuel assemblies containing 49 fuel rods each. During the first refueling, 166 of the 7 x 7 fuel assemblies were replaced with a like number of one water rod 8 x 8 fuel assemblies containing 63 fuel rods each. During the second refueling, an additional 156 of the original fuel assemblies were replaced with two water rod retrofit 8 x 8R fuel bundles containing 62 fuel rods each. During the third refueling outage, another 232 of the 7 x 7 fuel bundles were replaced with P 8 x 8 fuel assemblies, each containing 62 fuel rods. The prepressurized fuel assemblies (P 8 x 8R) are essentially identical from a core physics standpoint to the two water rod fuel assemblies (8 x 8R) except that they are prepressurized with about three rather than one atmospheres of helium to minimize fuel clad interaction. Our evaluation of the P 8 x 8R fuel is discussed in the safety evaluation attached to our letter of April 16, 1979 to General Electric approving the use of this fuel in BWR reload licensing applications. The larger inventory of helium gas improves the gap conductance between fuel pellets and cladding resulting in reductions in fuel temperatures, thermal expansion and fission gas release. The pressurized rods operate at effectively lower linear heat generation rates and are therefore expected to yield performance benefits in terms of fuel reliability. The increased prepressurization also results in improved margin to MAPLHGR limits by reducing stored energy.

During the current refueling outage, all of the remaining 214 original 7 x 7 fuel bundles will be replaced along with 46 of the 8 x 8 fuel assemblies. Thus, a total of 260 new fuel assemblies will be loaded in the core, consisting of 256 of the P8 x 8R fuel bundles and 4 lead test assemblies (two GLTA-1 and two GLTA-2). The four lead test assemblies (LTAs) are exactly the same as the standard P8DRB284L (P8 x 8R) reload bundle fuel except for a small axial section of increased Gadolinia content in some rods. Test measurements will be performed on these bundles during Cycle 5 to benchmark the effect of this increased Gadolinia content. All approved thermal-mechanical and reload methods described in NEDE-24011-P-A, "General Electric Standard Application for Reload," will hold for these LTAs.

With this refueling, Browns Ferry Unit 1 will continue to be on an 18 month refueling cycle. Units Nos. 2 and 3 are also on 18 month refueling cycles.

As noted above, this reload involves loading of prepressurized GE 8 x 8 retrofit (P8 x 8R) fuel. This is the same type of fuel as was loaded during the last reloads for all three Browns Ferry Units. The description of the nuclear and mechanical designs of 8 x 8 retrofit fuel is contained in References 3 and 4. Reference 3 also contains a complete set of references to topical reports which describe GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, and information regarding the applicability of these methods to cores containing a mixture of fuel. The use and safety implications of prepressurized fuel have been found acceptable per Reference 4. The conclusions of Reference 5, which was cited above, found that the methods of Reference 3 were generally applicable to prepressurized fuel. Therefore, unless otherwise specified, Reference 3, as supported by Reference 5, is adequate justification for the current application of prepressurized fuel.

3.0 Evaluation

3.1 Reactor Physics

The reload application follows the procedure described in NEDE-24011-P, "Generic Reload Fuel Application." We have reviewed this application and the consequent Technical Specification changes. The transient analysis input parameters are typical for BWRs and are acceptable. Core wide transient analysis results are given for the limiting transients and the required operating limit values for MCPR are given for each fuel type. The revised MCPR limits are required by the reload and they are acceptable.

3.2 Thermal Hydraulics

As stated in Reference 3, for BWR cores which reload with GE's retrofit 8x8R fuel, the safety limit minimum critical power ratio (SLMCPR) resulting from either core-wide or localized abnormal operational transients is equal to 1.07. When meeting this SLMCPR during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

To assure that the fuel cladding integrity SLMCPR will not be violated during any abnormal operational transient or fuel misloading, the most limiting events have been reanalyzed for this reload by the licensee, in order to determine which event results in the largest reduction in the minimum critical power ratio. These events have been analyzed for the exposed fuel and fresh fuel. Addition of the largest reductions in critical power ratio to the SLMCPR was used to establish the operating limits for each fuel type.

We have found the methods used for this analysis consistent with previously approved past practice (Reference 3). We have found the results of this analysis and the corresponding Technical Specification changes acceptable.

3.3 ECCS Appendix K

Input data and results for ECCS analysis have been given in References 1 and 2. The information presented fulfills the requirements for each analyses outlined in Reference 3.

We have reviewed the analyses and information submitted for the reload and conclude that BF-1 will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when it is operated in accordance with the Technical Specifications we are issuing with this amendment. Supplemental calculations that address the issues of NUREG-0630 have also been given in Reference 2.

3.4 Changes to Technical Specification

Our evaluation of the specific changes to the Technical Specifications resulting from the current reload is presented below:

Pgs. 9, 16, 131 and 160 - Since this reload removes the last of the original 7 x 7 fuel elements, the linear heat generation rate limit on these fuel elements is no longer pertinent and is being removed from the Technical Specifications and the bases.

Pgs. 19, 25 and 169 - This is the first reload for BF-1 in which the transients were analyzed by General Electric's ODYN Code as required by the staff. An additional citation is being added to the Technical Specifications to reference NRC's approval of this Code for core reloads.

Pgs. 19 and 221 - Section 2.1 of the Technical Specifications contains the bases for the "limiting safety system settings related to fuel cladding integrity." At the bottom of page 19 there is presently a paragraph relating to operation in the natural circulation mode. This paragraph is being moved, verbatim, to the bases for recirculation pump operation on p. 221, which is a more appropriate location. There is no safety significance to this reformating of the Technical Specifications.

Pgs. 30 and 219 - In Sections 2.2 (bases for reactor coolant system integrity) and 3.6.D/4.6.D (bases for relief valves), the value for the total capacity of the 13 relief valves is being increased from 82.6% to 83.9%. The value of 83.9 percent total relief capacity is derived from the values of 77.46 percent for 12 SRV's operable out of a total of 13 SRV's. The capacity of 77.46 percent of nuclear boiler rated steam flow, as listed in the BF 1 Reload 4 Supplemental Reload Licensing submittal, was calculated based on certified valve capacity for a 5.125-inch throat diameter valve (869,000 lbs/hour at 1,090 +3 psig) issued by the ASME National Board of Boiler and Pressure Vessel Inspectors. The certified values are obtained by testing and are listed as 90 percent of the measured capacity values for conservatism. The proposed change is supported by the reload submittal and is acceptable.

Pgs. 122, 123, 124 and 129 - As described in the discussion section of this safety evaluation, the reload for BF-1 will contain four LTAs. In order to obtain additional physics data, special cold criticality tests have been planned for this cycle. These criticality tests require suspension of the rod sequence control system (RSCS) constraints by means of the individual rod bypass switches. This testing is planned as part of the Lead Test Assembly program in which TVA and GE are participating. We have been kept appraised of this program through discussions and meetings, such as the meeting between TVA, GE and NRC staff in Bethesda, Md. on July 14, 1981. The next aspect of the program will include loading of four LTA's in the October 1981 refueling of Browns Ferry Unit No. 3. An analysis was performed to show that a postulated rod drop accident involving control rods withdrawn during the cold critical test would not exceed the peak fuel enthalpy design limit of 280 cal/gm. The rod worth minimizer (RWM) will be programmed to ensure adherence to the withdrawal sequence specified in the cold critical test procedure. The RWM must be operable for this test; a second licensed operator may not be used in lieu of the RWM for this testing. The proposed changes in the RSCS below 20% rated power - in conjunction with the compensatory measures - is acceptable.

Pgs. 143 and 145 - These changes are administrative changes that remove references to nonapplicable technical specification requirements. These changes do not affect any actual limiting conditions for operation; therefore, plant safety is not affected.

Pgs. 158 and 159 - As a result of previous changes to the Technical Specifications, sections 3.5.H and 4.5.H (Maintenance of Filled Discharge Pipe) is now located on two pages with half a page in between the lead sentence and the requirements. The proposed change is to relocate the parts of sections 3.5.H and 4.5.H now on page 159 to page 158 without any change in the wording. This reformating will improve clarity and has no safety significance.

Pgs. 159 and 169 - As supported by the reload submittal, the power spiking penalty is being removed from the linear heat generation rate (LHGR) limits for the 8 x 8, 8 x 8R and P8 x 8R fuel assemblies. This same change was previously made for BF-2 and BF-3 by Amendment No. 67 to Facility License No. DPR-52 on June 12, 1981 and by Amendment No. 37 to Facility License No. DPR-68 on January 12, 1981. The proposed change is supported by the reload submittal and is acceptable.

Pgs 160 and 172b - As supported by the reload submittal, the operating limit MCPR's are being changed. Since the MCPR's were determined by the ODYN Code (rather than the REDY Code), OLMCPR's are now calculated from two curves rather than being a single value (or a ramp change with fuel exposure).

Pg. 160a - Whenever the reactor power is equal to or greater than 25% thermal power, section 4.1.B of the Technical Specifications requires that the ratio of Fraction of Rated Power (FRP) to Core Maximum Fraction of Limiting Power Density (CMFLPD) shall be checked daily and the APRM scram trip setpoint(s) and the rod block trip setpoint (S_{RB}) recalculated and adjusted if the ratio is less than one (1). Unlike the BWR Standard Technical Specifications (NUREG-0123, Rev. 3), the Browns Ferry Technical Specifications do not provide a specified time to initiate corrective action or a time period to adjust the setpoints. Also, any excursion above this limit is now subject to the reporting requirements of Section 6.7.2.b(2). Under the old MCHFR correlations, the peaking factor (MFLPD/FRP) adjustment to the flow biased scram and rod block equations had relevance to maintaining core limits in certain flow excursion transients. Since adoption of CPR correlations, this is no longer the case and the flow biased equations now serve as a backup to the fixed (120%) scram and the RBM system, and provides additional conservatism for transients. Credit is not taken for the flow biased trips in the Browns Ferry transient analyses. Therefore, there is sufficient justification for relaxing the corrective action and time allowances in comparison to the standard core limits (MCPR, LHGR, etc.). Section 3.5.L is being modified to incorporate language similar to the BWR Standard Technical Specifications on the time permitted to initiate corrective action and to bring the factor within limits.

Pgs. 171, 172 and 172a - These revised pages present the new MAPLHGR versus average planar exposure limits determined by the supplemental reload analysis.

3.5 Plant Modifications

During this refueling outage, 67 significant modifications are being performed in addition to refueling, inservice inspection, surveillance and calibration tests, equipment overhaul and other maintenance performed during a refueling outage. These modifications are described in TVA's letter to us of May 22, 1981 and in the monthly operating reports. The most significant of the modifications are the torus integrity modifications being performed as part of the Mark I Containment Program. Another major modification is the changes being made to the BF-1 and BF-2 electrical systems; these electrical modifications are described and evaluated in a separate amendment.

3.5.1 Hydrogen Monitoring System

One of the modifications being performed, which requires changes to the Technical Specifications, is replacement of the containment hydrogen-oxygen monitoring system. This is the same type of monitoring system installed at the last refueling outages in Units 2 and 3. A complete description and evaluation of the new monitoring system is included in Amendment No. 58 to Facility Operating License No. DPR-52 issued November 12, 1980 and in Amendment No. 37 to Facility Operating License No. DPR-68 issued January 12, 1981 for Browns Ferry Unit Nos. 2 and 3, respectively. The evaluations contained therein are incorporated herein by reference. We conclude that the monitoring system meets the requirements in NUREG-0737 ("Clarification of TMI Action Plan Requirements") and that the proposed changes to the Technical Specifications are acceptable.

3.5.2 Torus Modifications

Numerous modifications are being implemented in the Unit 1 torus during the current refueling outage as part of the Mark I Containment Program. These modifications are required by NRC to restore the originally intended margins of safety in the containment design. The structural modifications to the torus containment include addition of torus tiedowns, addition of ring girder reinforcement and reinforcing attached piping nozzles. Vent System modifications include shortening the downcomers, adding local reinforcement to the vent header and adding new tie bars to the downcomers. Attached piping is being strengthened including modification of the ECCS header support. Many changes are being made to the safety relief valve piping system including adding quencher arms to the ramshead, adding quencher arm and ramshead supports, adding ten-inch vacuum valves, reinforcing the ring girder at the SRV hanger attachment, rerouting of piping and adding new snubbers and supports for the piping. These modifications have taken much longer to implement than originally estimated and have considerably extended the Unit 1 outage. When Unit 1 shutdown on April 11, 1981, the scheduled restart date was July 23, 1981. The projected startup date has slipped to about mid-September 1981 - almost two months longer than estimated.

The modifications to the torus and piping systems requires some changes to the Technical Specifications, as discussed below:

Pgs. 227 and 267 - The minimum torus water level limits in Section 3.7.A.1.a and in the bases for this Section are being changed from -7" (differential pressure control greater than 0 psid) to -6.25" and from -8" (0 psid differential pressure control) to -7.25" - a change in each case of 0.75". There are 15-inch by 15-inch sealed box beams being added as support for the safety relief valve lines, and HPCI-RCIC internal supports. Addition of these supports will result in appreciable water displacement. Calculations indicate that the box beams and HPCI-RCIC supports will increase the torus water level approximately 3/4 inch due to their presence. This rise in the torus water level is reflected in these revised technical specification values.

Pgs. 235a and 269 - In Section 3.7.A.6.a (and the bases therefore), the setpoint for the drywell-suppression chamber (wetwell) differential pressure control (ΔP) is being changed from 1.3 psid to 1.1 psid. Downcomer water clearing loads are greatly reduced by physically shortening the downcomers (by almost one foot) and imposing a drywell-wetwell ΔP. The Browns Ferry unique loads were determined by considering a differential pressure of 1.10 psid at the maximum allowable torus water level. In order to be consistent with this analysis the technical specification associated with the ΔP control has been established at 1.10 psid.

Pg. 268 - In the bases for the limits established for primary containment, there is a discussion of steam condensing loads associated with relief valve operation. The peak temperature of the torus water used in the evaluation is being changed from 160°F to 200°F local temperature.

During the current refuel outage the T-quenchers are being added to the safety-relief valve discharge device. The NRC licensed value for the T-quencher is 200°F local water temperature (to avoid excessive steam condensing loads). This technical specification change is needed to reflect that T-quencher licensed value of temperature.

3.5.3 Containment Purge System

In response to our generic letters of September 27, 1979 and October 22, 1979 to "All Light Water Reactors," TVA is modifying the containment purge system for Unit 1 during this outage to satisfy applicable requirements of NRC Branch Technical Position CSB 6-4 regarding valve closure times and addition of debris screens. Table 3.7.A (pages 251 and 252) is being revised to reflect the significant reduction in the maximum allowable operating time for the purge valves. On the nitrogen purge valves the operating time is being reduced from 10 seconds to 5 seconds and on the purge inlet and isolation valves the operating time is being reduced from 90 seconds to only 2.5 seconds. The faster valve closure times significantly reduce potential offsite doses. The addition of the debris screens provides protection against foreign material entering the purge ducting and interfering with closure of the purge valves. In their letter of June 2, 1981, TVA provided the data and analysis to demonstrate that the purge valves are adequate for closure against the design basis loss-of-coolant accident forces. We have concluded that the plant modifications and changes to the Technical Specifications are significant improvements in plant safety and should be approved.

3.5.4 HPCI Bypass Valve

During this refueling outage, a one-inch bypass valve is being added around the HPCI steam supply outboard isolation valve, FCV73-3. During quarterly surveillance testing on HPCI isolation valve FCV 73-3, in which the valve is closed and reopened, the steamline downstream from FCV 73-3 is subjected to thermal stresses from the closure and subsequent reopening. Addition of FCV 73-81 will relieve those stresses. This is a one-inch valve. It is an isolation group 4 valve with a maximum closing time of 10 seconds. Since this is an isolation valve, it is being added to the list of valves in Tables 3.7.A (p. 251) and 3.7.D (p. 260).

4.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

5.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: September 15, 1981

References

1. Supplemental Reload Licensing Submittal for Browns Ferry Nuclear Plant Unit 1, Reload No. 4 (Cycle 5) Y1003J01A19 dated March 1981.

2. Errata and Addenda Sheet No. 2 dated April 1981 to NEDO-24056, "Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 1 issued September 1977.

3. "General Electric Boiling Water Reactor Generic Reload Application," NEDE-24011-P-A, May 1977.

4. Letter, R. E. Engel (GE) to U. S. Nuclear Regulatory Commission, dated January 30, 1979.

5. Letter, T. A. Ippolito (USNRC) to R. Gridley (GE), April 16, 1979, and enclosed SER.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-259

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 76 to Facility Operating License No. DPR-33 issued to Tennessee Valley Authority (the licensee), which revised the Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit No. 1 (the facility) located in Limestone County, Alabama. The amendment is effective as of the date of issuance.

This amendment (1) incorporates the limiting conditions for operation of the facility in the fifth fuel cycle following the current refueling outage, (2) reflects new primary containment atmospheric monitoring instrumentation installed during this outage and (3) reflects modifications which the Commission required to be made to the torus.

The application for this amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated April 29, 1981, as supplemented by letters dated June 12, 1981 and July 13, 1981, (2) Amendment No. 76 to License No. DPR-33, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 15th day of September 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

T. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing