



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

TENNESSEE VALLEY AUTHORITY  
DOCKET NO. 50-260  
BROWNS FERRY NUCLEAR PLANT, UNIT 2  
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 185  
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 6, 1990 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
  - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
  - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

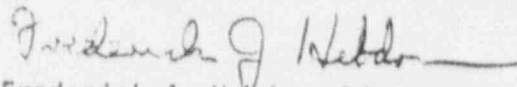
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-52 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 185, 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 its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Frederick J. Heddon, Director  
Project Directorate II-4, NRR  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: January 9, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 185

FACILITY OPERATING LICENSE NO. DPR-52

LOCKET NO. 50-260

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Overleaf pages\* are provided to maintain document completeness.

<u>REMOVE</u>	<u>INSERT</u>
3.2/4.2-14	3.2/4.2-14
3.2/4.2-15	3.2/4.2-15
3.2/4.2-16	3.2/4.2-16*
3.2/4.2-17	3.2/4.2-17
3.2/4.2-22	3.2/4.2-22*
--	3.2/4.2-22a
3.2/4.2-44	3.2/4.2-44*
3.2/4.2-45	3.2/4.2-45
3.2/4.2-65	3.2/4.2-65*
3.2/4.2-66	3.2/4.2-66
3.5/4.5-16	3.5/4.5-16
3.5/4.5-17	3.5/4.5-17
3.5/4.5-28	3.5/4.5-28*
3.5/4.5-29	3.5/4.5-29
3.5/4.5-30	3.5/4.5-30
3.5/4.5-31	3.5/4.5-31*

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

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Below trip setting initiated HPCI.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 398" above vessel zero.	A	1. Below trip setting initiates CSS.  Multiplier relays initiate LPCI.  2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 398" above vessel zero.	A	1. Below trip settings, in conjunction with drywell high pressure, low water level permissive, ADS timer timed out and CSS or RHR pump running, initiates ADS.  2. Below trip settings, in conjunction with low reactor water level permissive, ADS timer timed out, ADS high drywell pressure bypass timer timed out, CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184, 185)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LIS-3-52 and LIS-3-62A)	312 5/16" above vessel zero. A (3 core height)	A	1. Below trip setting ; inadvertent operat containment spray ; accident conditio

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TABLE 3.2.B (Continued)

RFN Unit	Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	2	Instrument Channel - Drywell High Pressure (PIS-64-58 E-H)	$1 \leq p \leq 2.5$ psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.
	2	Instrument Channel - Drywell High Pressure (PIS-64-58 A-0)	$\leq 2.5$ psig	A	1. Above trip setting in conjunction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal. (15)
	2	Instrument Channel - Drywell High Pressure (PIS-64-58A-0)	$\leq 2.5$ psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
	2(16)	Instrument Channel - Drywell High Pressure (PIS-64-57A-0)	$\leq 2.5$ psig	A	1. Above trip setting, in conjunction with low reactor water level, low reactor water level permissive, ADS timer timed out, and CSS or RHR pump running, initiates ADS.

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TABLE 3.2.8 (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Pressure (PIS-3-74 A & B) (PIS-68-95, 96)	450 psig ± 15	A	1. Below trip setting permissive for opening CSS and LPCI admission valves.
2	Instrument Channel - Reactor Low Pressure (PS-3-74 A & B) (PS-68-95, 96)	230 psig ± 15	A	1. Recirculation discharge valve actuation.
2	Core Spray Auto Sequencing Timers (5)	6 ≤ t ≤ 8 sec.	B	1. With diesel power 2. One per motor
2	LPCI Auto Sequencing Timers (5)	0 ≤ t ≤ 1 sec.	B	1. With diesel power 2. One per motor
1	RHR SW A1, B3, C1, and D3 Timers	13 ≤ t ≤ 15 sec.	A	1. With diesel power 2. One per pump
2	Core Spray and LPCI Auto Sequencing Timers (6)	0 ≤ t ≤ 1 sec. 6 ≤ t ≤ 8 sec. 12 ≤ t ≤ 16 sec. 18 ≤ t ≤ 24 sec.	B	1. With normal power 2. One per CSS motor 3. Two per RHR motor
1	RHR SW A1, B3, C1, and D3 Timers	27 ≤ t ≤ 29 sec.	A	1. With normal power 2. One per pump

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TABLE 3.2.B (Continued)

BFN Unit	Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	2	Instrument Channel - RHR Discharge Pressure	100 $\pm$ 10 psig	A	1. Below trip setting defers ADS actuation.
	2	Instrument Channel CSS Pump Discharge Pressure	185 $\pm$ 10 psig	A	1. Below trip setting defers ADS actuation.
	1(3)	Core Spray Sparger to Reactor Pressure Vessel d/p	2 psid $\pm$ 0.4	A	1. Alarm to detect core sparger pipe break.
		RHR (LPCI) Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
	1	Core Spray Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
	1	ADS Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems and valves.

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TABLE 3.2.B (Continued)

Function	Trip Level Setting	Action	Remarks
Instrument Channel - Thermostat (Core Spray Area Cooler Fan)	≤ 100°F	A	1. Above trip setting starts Core Spray area cooler fans.
RHR Area Cooler Fan Logic	N/A	A	
Core Spray Area Cooler Fan Logic	N/A	A	
Instrument Channel - Core Spray Motors A or D Start	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
Instrument Channel - Core Spray Motor B or C Start	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
Instrument Channel - Core Spray Loop 1 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
Instrument Channel - Core Spray Loop 2 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
RHRSW Initiate Logic	N/A	(14)	
RPT Logic	N/A	(17)	1. Trips recirculation pumps on turbine control valve fast closure or stop valve closure > 30% power.

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3.2/4.2-22



TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
1(16)	ADS Timer	$t \leq 115$ sec.	A	1. Above trip setting in conjunction with low reactor water level permissive, low reactor water level; high drywell pressure or ADS high drywell pressure bypass timer timed out, and RHR or CSS pumps running, initiates ADS.
1(16)	ADS High Drywell Pressure Bypass Timer	$t \leq 322$ sec.	A	1. Above trip setting, in conjunction with low reactor water level permissive, low reactor water level, ADS timer timed out and RHR or CSS pumps running, initiates ADS.

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

Function	Functional Test	Calibration	Instrument Check
Instrument Channel Reactor Low Water Level (LIS-3-58A-0)	(1) (27)	Once/18 Months (28)	Once/day
Instrument Channel Reactor Low Water Level (LIS-3-184 & 185)	(1) (27)	Once/18 Months (28)	Once/day
Instrument Channel Reactor Low Water Level (LIS-3-52 & 62A)	(1) (27)	Once/18 Months (28)	Once/day
Instrument Channel Drywell High Pressure (PIS-64-58E-H)	(1) (27)	Once/18 Months (28)	none
Instrument Channel Drywell High Pressure (PIS-64-58A-0)	(1) (27)	Once/18 Months (28)	none
Instrument Channel Drywell High Pressure (PIS-64-57A-0)	(1) (27)	Once/18 Months (28)	none
Instrument Channel Reactor Low Pressure (PIS-3-74A&B, PS-3-74A&B) (PIS-68-95, PS-68-95) (PIS-58-96, PS-68-96)	(1) (27)	Once/6 Months (28)	none

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3.2/4.2-44

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TABLE 4.2.B (Continued)  
 SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
Core Spray Auto Sequencing Timers (Normal Power)	(4)	Once/operating cycle	none
Core Spray Auto Sequencing Timers (Diesel Power)	(4)	Once/operating cycle	none
LPCI Auto Sequencing Timers (Normal Power)	(4)	Once/operating cycle	none
LPCI Auto Sequencing Timers (Diesel Power)	(4)	Once/operating cycle	none
RHRSW A1, B3, C1, D3 Timers (Normal Power)	(4)	Once/operating cycle	none
RHRSW A1, B3, C1, D3 Timers (Diesel Power)	(4)	Once/operating cycle	none
ADS Timer	(4)	Once/operating cycle	none
ADS High Drywell Pressure Bypass Timer	(4)	Once/operating cycle	none

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### 3.2 BASES

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

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

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

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

The low water level instrumentation set to trip at 538 inches above vessel zero closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Groups 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 470 inches above vessel zero (Table 3.2.B) trips the recirculation pumps and initiates the RCIC and HPCI systems. The RCIC and HPCI system initiation opens the turbine steam supply valve which in turn initiates closure of the respective drain valves (Group 7).

The low water level instrumentation set to trip at  $\geq 398$  inches above vessel zero (Table 3.2.A) closes the Main Steam Isolation Valves, the Main Steam Line Drain Valves, and the Reactor Water Sample Valves (Group 1). Details of valve grouping and required closing times are given in Specification 3.7. These trip settings are adequate to prevent core uncovering in the case of a break in the largest line assuming the maximum closing time.

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

### 3.2 BASES (Cont'd)

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

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

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

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

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

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

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

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.G Automatic Depressurization System (ADS)

1. Six valves of the Automatic Depressurization System shall be OPERABLE:
  - (1) PRIOR TO STARTUP from a COLD CONDITION, or,
  - (2) whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 10 psig, except as specified in 3.5.G.2 and 3.5.G.3 below.
2. With one of the above required ADS valves inoperable, provided the HPCI system, the core spray system, and the LPCI system are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least a HOT SHUTDOWN CONDITION within the next 12 hours and reduce reactor steam dome pressure to  $\leq 105$  psig within 24 hours.
3. With two or more of the above required ADS valves inoperable, be in at least a HOT SHUTDOWN CONDITION within 12 hours and reduce reactor steam dome pressure to  $\leq 105$  psig within 24 hours.

4.5.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. No additional surveillances are required.



3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.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.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 and prior to the testing of the RHRS (LPCI and Containment Spray) and core spray system, the discharge piping of these systems shall be vented from the high point and water flow determined.
2. Following any period where the LPCI or core spray systems have not been required to be OPERABLE, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.
3. Whenever the HPCI or RCIC system is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.
4. When the RHRS and the CSS are required to be OPERABLE, the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure recorded.



3.5.E. High Pressure Coolant Injection System (HPCI)

The HPCIS is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCIS continues to operate until reactor vessel pressure is below the pressure at which LPCI operation or Core Spray system operation maintains core cooling. The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to pump 5000 gpm at reactor pressures between 1120 and 150 psig. The HPCIS is not required to be operable below 150 psig since this is well within the range of the low pressure cooling systems and below the pressure of any events for which HPCI is required to provide core cooling.

The HPCIS is not designed to operate at full capacity until reactor pressure exceeds 150 psig and the steam supply to the HPCI turbine is automatically isolated before reactor pressure decreases below 100 psig. The ADS, CSS, and RHRS (LPCI) must be OPERABLE when starting up from a Cold Condition. Steam pressure is sufficient at 150 psig to run the HPCI turbine for operability testing yet still below the shutoff head of the CSS and RHRS pumps so they will inject water into the vessel if required. The ADS provides additional backup to reduce pressure to the range where the CSS and RHRS will inject into the vessel if necessary. Considering the low reactor pressure, the redundancy and availability of CSS, RHRS, and ADS during startup from a Cold Condition, twelve hours is allowed as a reasonable time to demonstrate HPCI operability once sufficient steam pressure becomes available. The alternative to demonstrate HPCI operability prior to startup using auxiliary steam is provided for plant operating flexibility.

With the HPCIS inoperable, a seven-day period to return the system to service is justified based on the availability of the ADS, CSS, RHRS (LPCI) and the RCICS. The availability of these redundant and diversified systems provides adequate assurance of core cooling while HPCIS is out of service.

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

### 3.5 BASES (Cont'd)

#### 3.5.F Reactor Core Isolation Cooling System (RCICS)

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

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

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

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

#### 3.5.G Automatic Depressurization System (ADS)

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

Specification 3.6.D specifies the requirements for the pressure relief function of the valves. It is possible for any number of the valves assigned to the ADS to be incapable of performing their ADS functions because of instrumentation failures, yet be fully capable of performing their pressure relief function.

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

### 3.5 BASES (Cont'd)

valves to be operable, additional conservatism is provided to account for the possibility of a single failure in the ADS system.

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

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

#### 3.5.H. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI, HPCIS, and RCIGS are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in an OPERABLE condition. If a discharge pipe is not filled, the pumps that supply that line must be assumed to be inoperable for Technical Specification purposes.

The core spray and RHR system discharge piping high point vent is visually checked for water flow once a month and prior to testing to ensure that the lines are filled. The visual checking will avoid starting the core spray or RHR system with a discharge line not filled. In addition to the visual observation and to ensure a filled discharge line other than prior to testing, a pressure suppression chamber head tank is located approximately 20 feet above the discharge line high point to supply makeup water for these systems. The condensate head tank located approximately 100 feet above the discharge high point serves as a backup charging system when the pressure suppression chamber head tank is not in service. System discharge pressure indicators are used to determine the water level above the discharge line high point. The indicators will reflect approximately 30 psig for a water level at the high point and 45 psig for a water level in the pressure suppression chamber head tank and are monitored daily to ensure that the discharge lines are filled.

When in their normal standby condition, the suction for the HPCI and RCIC pumps are aligned to the condensate storage tank, which is physically at a higher elevation than the HPCIS and RCIGS piping. This assures that the HPCI and RCIC discharge piping remains filled. Further assurance is provided by observing water flow from these systems' high points monthly.

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

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

### 3.5 BASES (Cont'd)

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than  $\pm 20^\circ\text{F}$  relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting value for MAPLHGR is shown in Tables 3.5.I-1, 2, 3, and 4. The analyses supporting these limiting values are presented in Reference 1.

#### 3.5.J. Linear Heat Generation Rate (LHGR)

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

The LHGR shall be checked daily during reactor operation at  $\geq 25$  percent power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 percent rated thermal power, the R factor would have to be less than 0.241 which is precluded by a considerable margin when employing any permissible control rod pattern.

#### 3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

#### 3.5.L. APRM Setpoints

Operation is constrained to a maximum LHGR of 13.4 kW/ft for 8x8 fuel. This limit is reached when core maximum fraction of limiting power density (CMFLPD) equals 1.0. For the case where CMFLPD exceeds the fraction of rated thermal power, operation is permitted only at less than 100-percent rated power and only with APRM scram settings as required by Specification 3.5.L.1. The scram trip setting and rod block trip setting are adjusted to ensure that no combination