ENCLOSURE 4

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
NRC DOCKET NOS. 50-325 AND 50-324
OPERATING LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
ENHANCED OPTION I-A STABILITY TECHNICAL SPECIFICATIONS

MARKED-UP TECHNICAL SPECIFICATION AND BASES PAGES - UNIT 1

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MARKED-UP TECHNICAL SPECIFICATION AND BASES PAGES - UNIT 1

ITS Table 3.3.1.1-1

Changed per 96TSB02, ITS

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux - High[a]	≤ 120 divisions of full scale	≤ 120 divisions of full scale
2. Average Power Range Monitor		
a. Neutron Flux - High, 15%[b]	≤ 15% of RATED THERMAL POWER	≤ 15% of RATED THERMAL POWER
b. Flow-Biased Simulated Thermal Power - High[c][d]	≤ (0.66W + 64%) with a maximum ≤ 113.5% of RATED THERMAL POWER	~~≤ (0.66W + 67%) with a maximum ≤ 115.5% of RATED THERMAL POWER~~
c. Fixed Neutron Flux - High[d]	≤ 120% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1045 psig	≤ 1045 psig
4. Reactor Vessel Water Level - Low, Level 1	≥ +162.5 inches[g]	≥ +162.5 inches[g]
5. Main Steam Line Isolation Valve - Closure[e]	≤ 10% closed	≤ 10% closed
6. (Deleted)		
7. Drywell Pressure - High	≤ 2 psig	≤ 2 psig
8. Scram Discharge Volume Water Level - High	≤ 109 gallons	≤ 109 gallons
9. Turbine Stop Valve - Closure[f]	≤ 10% closed	≤ 10% closed
10. Turbine Control Valve Fast Closure, Control Oil Pressure - Low[f]	≥ 500 psig	≥ 500 psig

Note (b): Allowable Value specified in the COLR

Changed per 96TSB02, ITS

Changed per 96TSB02, ITS

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

NOTES

(a) The Intermediate Range Monitor scram functions are automatically bypassed when the reactor mode switch is placed in the Run position and the Average Power Range Monitors are on scale.

(b) This Average Power Range Monitor scram function is a fixed point and is increased when the reactor mode switch is placed in the Run position.

(c) The Average Power Range Monitor scram function is varied, Figure 2.2.1-1, as a function of the fraction of rated recirculation loop flow (W) in percent.

ITS SR 3.3.1.1.14 (d) The APRM flow-biased simulated thermal power signal is fed through a time constant circuit of ~~approximately 6~~ ≤7 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.

(e) The Main Steam Line Isolation Valve-Closure scram function is automatically bypassed when the reactor mode switch is in other than the Run position.

(f) These scram functions are bypassed when THERMAL POWER is less than 30% of RATED THERMAL POWER as measured by turbine first stage pressure.

(g) Vessel water levels refer to REFERENCE LEVEL ZERO.

Changed per 96TSB02, ITS

Changed per 96TSB02, ITS

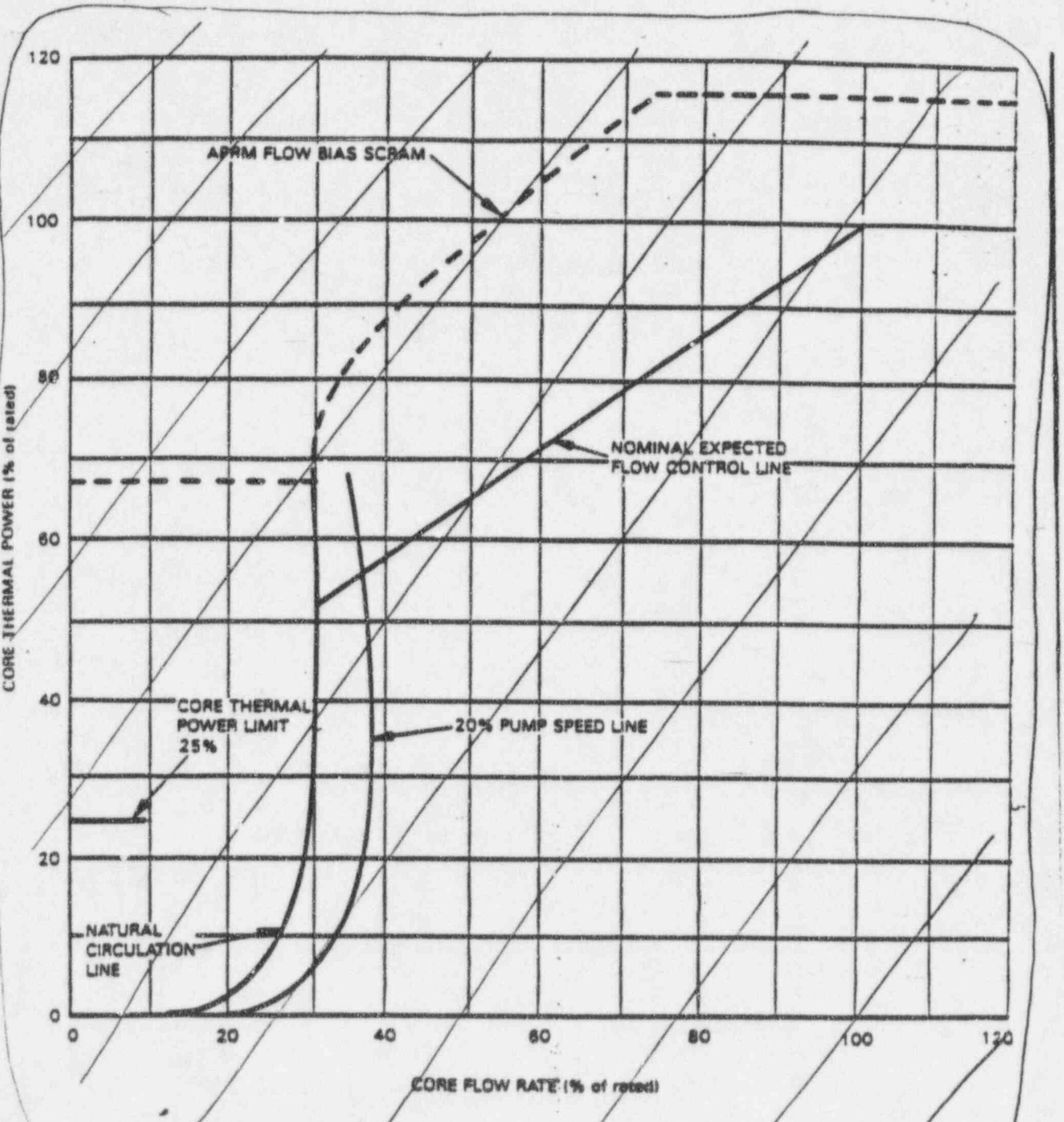


Figure 2.2.1-1. APRM Flow Bias Scram Relationship to Normal Operating Conditions

ITS 3.3.1.1 and Table 3.3.1.1-1

Changed per 96TSB02, ITS

TABLE 4.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	D	S/U[b][c], W[c]	R	2
	D	W	R	3, 4, 5
b. Inoperative	NA	W[d]	NA	2, 3, 4, 5
2. Average Power Range Monitor:				
a. Neutron Flux - High 15%	S	S/U[b][m], W[d]	Q	2
	S	W[n]	Q	5
b. Flow-Biased Simulated Thermal Power - High	S	S/U[b], Q	W[e], Q	1
c. Fixed Neutron Flux - High, 120%	S	S/U[b], Q	W[e], Q	1
d. Inoperative	NA	Q[m][n]	NA	1, 2, 5
e. Downscale	NA	Q	NA	1
f. LPRM	D	NA	(g)	1, 2, 5
3. Reactor Vessel Steam Dome Pressure - High				
Transmitter:	NA[k]	NA	R[l]	1, 2
Trip Logic:	D	Q	Q	1, 2
4. Reactor Vessel Water Level - Low, Level 1				
Transmitter:	NA[k]	NA	R[l]	1, 2
Trip Logic:	D	Q	Q	1, 2

Add SR 3.3.1.1.18

Add Note 3 to SR 3.3.1.1.11

Changed per 96TSB02, ITS

Changed per 96TSB02, ITS

TABLE 4.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Within 24 hours prior to startup, if not performed within the previous 7 days.

(c) The IRM channels shall be compared to the APRM channels and the SRM instruments for overlap during each startup, if not performed within the previous 7 days.

(d) When changing from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, perform the required surveillance within 12 hours after entering OPERATIONAL CONDITION 2, if not performed within the previous 7 days.

(e) This calibration shall consist of the adjustment of the APRM readout to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

(f) ~~This calibration shall consist of the adjustment of the APRM flow-biased simulated thermal power channel to conform to a calibrated flow signal.~~

(g) The LPRMs shall be calibrated at least once per effective full power month (EFPM) using the TIP system.

(h) This calibration shall consist of a physical inspection and actuation of these position switches.

(i) (Deleted)

(j) (Deleted)

(k) The transmitter channel check is satisfied by the trip unit channel check. A separate transmitter check is not required.

(l) Transmitters are exempted from the quarterly channel calibration.

(m) Placement of Reactor Mode Switch into the Startup/Hot Standby position is permitted for the purpose of performing the required surveillance prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

(n) Placement of Reactor Mode Switch into the Shutdown or Refuel position is permitted for the purpose of performing the required surveillance provided all control rods are fully inserted and the vessel head bolts are tensioned.

(o) Surveillance is not required when THERMAL POWER is less than 30% of RATED THERMAL POWER.

Changed per 96TSB02, ITS

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

Changed per 96TSB02, ITS

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant recirculation loops shall be in operation with the cross-tie valve closed, the pump discharge valves OPERABLE, and the pump discharge bypass valves OPERABLE or closed and

a. Total core flow shall be greater than or equal to 35 million lbs/hr, or

b. THERMAL POWER shall be less than or equal to the limit specified in Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

Changed per 96TSB02, ITS

ACTION:

a. With both reactor coolant system recirculation loops not in operation, immediately initiate an orderly reduction of THERMAL POWER so that it is less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours, and restore both loops to operation within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. With one reactor coolant system recirculation loop not in operation, immediately initiate either an orderly reduction of THERMAL POWER so that it is less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours or increase core flow so that it is greater than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours, and restore both loops to operation within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

Changed per 96TSB02, ITS

c. With two reactor coolant system recirculation loops in operation and total core flow less than 35 million lbs/hr and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1:

   1. Immediately initiate action to reduce THERMAL POWER so that it is less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours, or

   2. Immediately initiate action to increase core flow so that it is greater than 35 million lbs/hr within 2 hours, or

   3. Determine the APRM and LPRM neutron flux noise levels within 2 hours, and:

      a) If the APRM and LPRM neutron flux noise levels are less than three times their established baseline levels or less than 5% peak-to-peak, continue to determine the noise levels at least once per 24 hours and within 1 hour after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER, or

      b) If the APRM or LPRM neutron flux noise levels are greater than or equal to three times their established baseline levels and greater than 5% peak-to-peak, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours by increasing core flow to greater than 35 million lbs/hr and/or by initiating an orderly reduction of

*See Special Test Exception 3.10.4.

Changed per 96TSB02, ITS

Changed per 96TSB02, ITS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1.

Changed per 96TSB02, ITS

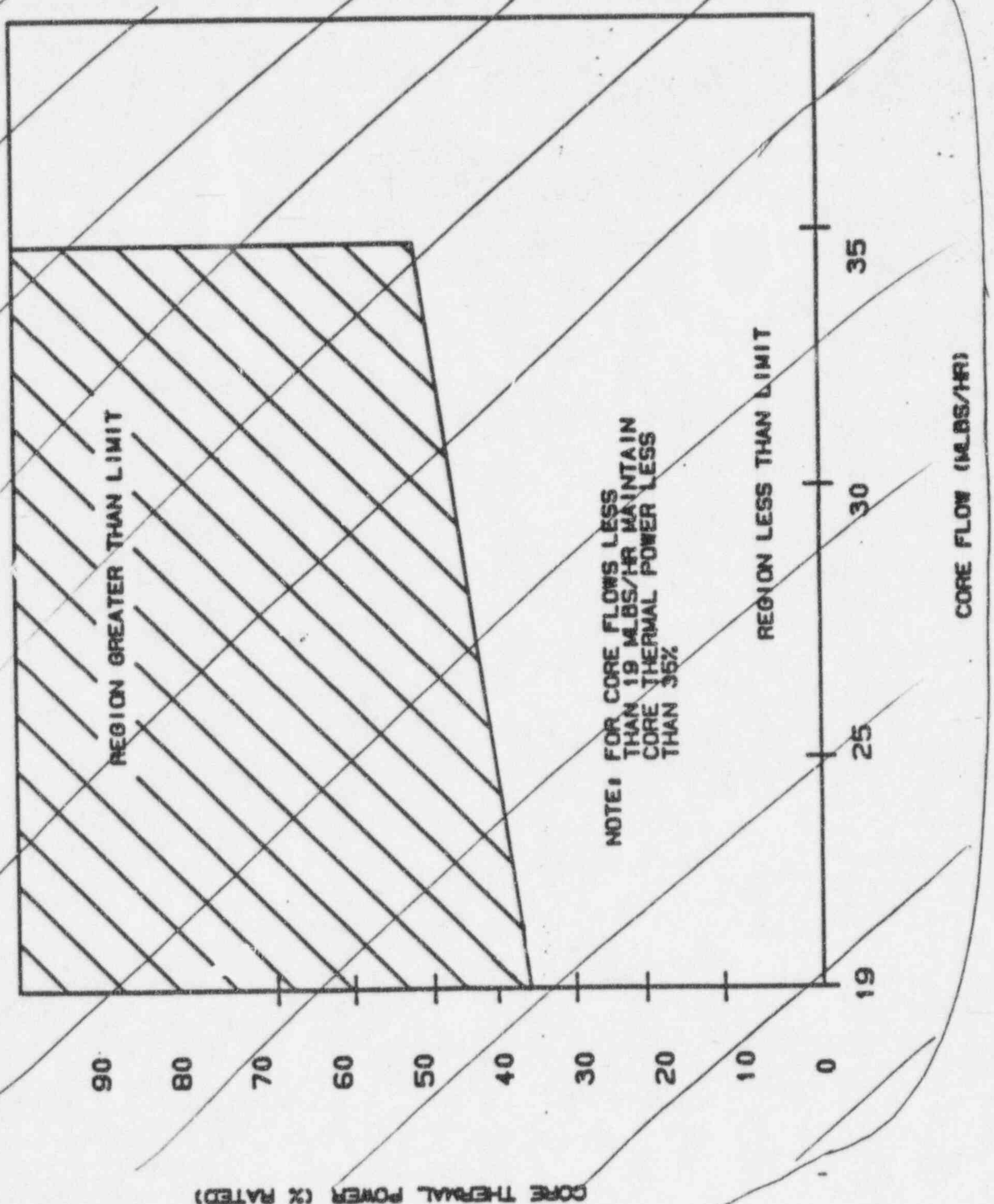
SURVEILLANCE REQUIREMENTS

4.4.1.1.1 Each pump discharge valve and bypass valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each COLD SHUTDOWN which exceeds 48 hours, if not performed in the previous 31 days.

4.4.1.1.2 Each pump discharge bypass valve, if not OPERABLE, shall be verified to be closed at least once per 31 days.

~~4.4.1.1.3 Establish baseline APRM and LPRM neutron flux noise values at a point below the 100% rated rod line during startup testing following each refueling outage.~~

FIGURE [illegible].1.1-1
THERMAL POWER LIMITATIONS



ITS 5.6.5

5.6.5.a 3 The Allowable Value for Function 2.b, APRM Flow Biased Simulated Thermal Power - High, for Specification 3.3.1.1; and

Changed per 96TSB02, ITS

CORE OPERATING LIMITS REPORT (Continued)

b. The core flow and core power adjustments for Specification 3.2.2.1

c. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specifications 3.2.2.1 and 3.2.2.2.

d. The rod block monitor upscale trip setpoint and allowable value for Specification 3.3.4.

and shall be documented in the CORE OPERATING LIMITS REPORT.

6.9.3.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

Changed per 96TSB02, ITS

a. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).

b. The May 18, 1984 and October 22, 1984 NRC Safety Evaluation Reports for the Brunswick Reload Methodologies described in:

1. Topical Report NF-1583.01, "A Description and Validation of Steady-State Analysis Methods for Boiling Water Reactors," February 1983.

Insert 5.6.5.b

2. Topical Report NF-1583.02, "Methods of RECORD," February 1983.

3. Topical Report NF-1583.03, "Methods of PRESTO-B," February 1983.

Changed per 96TSB02 ITS

4. Topical Report NF-1583.04, "Verification of CP&L Reference BWR Thermal-Hydraulic Methods Using the FIBWR Code," May 1983.

6.9.3.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

6.9.3.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.10 RECORD RETENTION

Facility records shall be retained in accordance with ANSI-N45.2.9-1974.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

Insert 5.6.5.b

2. NEDO-32339-A, "Reactor Stability LongTerm Solution: Enhanced Option I-A," July 1995.

3. NEDC-32339-P Supplement 1, "Reactor Stability Long Term Solution: Enhanced Option I-A ODYSY Computer Code," March 1994 (Approved in NRC Safety Evaluation dated January 4, 1996).

4. NEDO-32339 Supplement 3, "Reactor Stability Long Term Solution: Enhanced Option I-A Flow Mapping Methodology," August 1995 (Approved in NRC Safety Evaluation dated May 28, 1996).

Changed per 96TSB02, ITS

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation for longer than 24 hours with a reactor core coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, evaluated, and determined to be acceptable.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does present a hazard in case of a design basis accident by increasing the blowdown area and eliminating the capability of reflooding the core--thus, the requirement for shutdown of the facility with a jet pump inoperable.

The established characteristics for the criteria of 4.4.1.2 are bands of values that encompass the normal scatter of the data. The scatter in the data can be attributed primarily to monitoring inaccuracies and instrumentation tolerances. An evaluation of these factors will be used to determine the widths of the bands. The bands will be centered about the expected values determined from operating data. The acceptance criteria will be these bands plus the appropriate percentage of the expected values at any point. The acceptance criteria will be updated as required to reflect any changes in the recirculation system that would affect the monitored variables.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures should be within 50°F of each other prior to start-up of an idle loop.

Since the coolant in the bottom of the vessel is at a lower temperature than the water in the upper regions of the core, undue stress on the vessel would result if the temperature difference were greater than 145°F.

Neutron flux noise limits are established to ensure early detection of limit cycle neutron flux oscillations. BWR cores typically operate with neutron flux noise caused by random boiling and flow noise. Typical neutron flux noise levels of 1 to 12% of rated power (peak-to-peak) have been reported for the range of low to high recirculation loop flow during both single and dual recirculation loop operation. Neutron flux noise levels significantly larger than these values are considered in the thermal/mechanical fuel design and are found to be of negligible consequence. In addition, stability tests at operating BWR's have demonstrated that when stability related neutron flux limit cycle oscillations occur they result in peak-to-peak neutron flux limit cycles 5 to 10 times the typical values. Therefore, actions taken to reduce neutron flux noise levels exceeding three (3) times the typical value are sufficient to ensure early detection of limit cycle neutron flux oscillations.

Data to establish baseline APRM and LPRM neutron flux noise values is obtained at a point below the 100% rated rod line. A minimum of two detectors of one LPRM string per core octant and two detectors of one LPRM string near the center of the core should be monitored. Detectors used for monitoring should be selected to provide core wide representation. Substitutions are permitted for inoperable LPRM detectors.

Changed per 96TSB02, ITS

~~These specifications are based on the guidance of General Electric SIL #380, Rev. 1, 2-10-84.~~

3/4.4.2 SAFETY/RELIEF VALVES

The reactor coolant system safety valve function of the safety-relief valves operate to prevent the system from being pressurized above the Safety Limit of 1325 psig. The system is designed to meet the requirements of the ASME Boiler and Pressure Vessel Code Section III for the pressure vessel and ANSI B31.1, 1975, Code for the reactor coolant system piping.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shut down to allow further investigation and corrective action. Monitoring leakage at eight hour intervals is in conformance with the 12/21/89 NRC SER for GL 88-01.

3/4.4.4 CHEMISTRY

The reactor water chemistry limits are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low; thus, the higher limit on chlorides is permitted during full power operation. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides, and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity outside the limits, additional samples must be examined to ensure that the chlorides are not exceeding the limits.

Changed per 96TSB02, ITS

Insert New Specification 3.2.3 and B 3.2.3

Insert New Specification 3.2.3, "Fraction of Core Boiling Boundary (FCBB)," and associated Bases as shown in Enclosures 6 and 7 for BNP Units 1 and 2, respectively.

Insert New Specification 3.3.1.3 and B 3.3.1.3

Insert New Specification 3.3.1.3, "Period Based Detection System (PBDS)," and associated Bases as shown in Enclosures 6 and 7 for BNP Units 1 and 2, respectively.

ENCLOSURE 5

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
NRC DOCKET NOS. 50-325 AND 50-324
OPERATING LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
ENHANCED OPTION I-A STABILITY TECHNICAL SPECIFICATIONS

MARKED-UP TECHNICAL SPECIFICATION AND BASES PAGES - UNIT 2

ITS Table 3.3.1.1-1

Changed per 96TSB02, ITS

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux - High [a]	≤ 120 divisions of full scale	≤ 120 divisions of full scale
2. Average Power Range Monitor		
a. Neutron Flux - High, 15% [b]	≤ 15% of RATED THERMAL POWER	≤ 15% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power - High [c][d]	≤ (0.66 W + 64%) with a maximum ≤ 113.5% of RATED THERMAL POWER	~~≤ (0.66 W + 67%) with a maximum ≤ 115.5% of RATED THERMAL POWER~~ Note(b): Allowable Value specified in the COLR
c. Fixed Neutron Flux - High [d]	≤ 120% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ 1045 psig	≤ 1045 psig
4. Reactor Vessel Water Level - Low, Level 1	≥ +162.5 inches [g]	≥ +162.5 inches [g]
5. Main Steam Line Isolation Valve - Closure [e]	≤ 10% closed	≤ 10% closed
6. (Deleted)		
7. Drywell Pressure - High	≤ 2 psig	≤ 2 psig
8. Scram Discharge Volume Water Level - High	≤ 109 gallons	≤ 109 gallons
9. Turbine Stop Valve-Closure [f]	≤ 10% closed	≤ 10% closed
10. Turbine Control Valve Fast Closure, Control Oil Pressure-Low [f]	≥ 500 psig	≥ 500 psig

Changed per 96TSB02, ITS

Changed per 96TSB02, ITS

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

NOTES

(a) The Intermediate Range Monitor scram functions are automatically bypassed when the reactor mode switch is placed in the Run position and the Average Power Range Monitors are on scale.

(b) This Average Power Range Monitor scram function is a fixed point and is increased when the reactor mode switch is placed in the Run position.

(c) The Average Power Range Monitor scram function is varied, Figure 2.2.1-1, ~~as a function of the fraction of rated recirculation loop flow~~ (W) in percent.

ITS SR 3.3.1.1.14 (d) The APRM flow-biased simulated thermal power signal is fed through a time constant circuit of ~~approximately 6~~ ≤7 seconds. ~~The APRM fixed high neutron flux signal does not incorporate the time constant~~, but responds directly to instantaneous neutron flux.

(e) The Main Steam Line Isolation Valve-Closure scram function is automatically bypassed when the reactor mode switch is in other than the Run position.

(f) These scram functions are bypassed when THERMAL POWER is less than 30% of RATED THERMAL POWER as measured by turbine first stage pressure.

(g) Vessel water levels refer to REFERENCE LEVEL ZERO.

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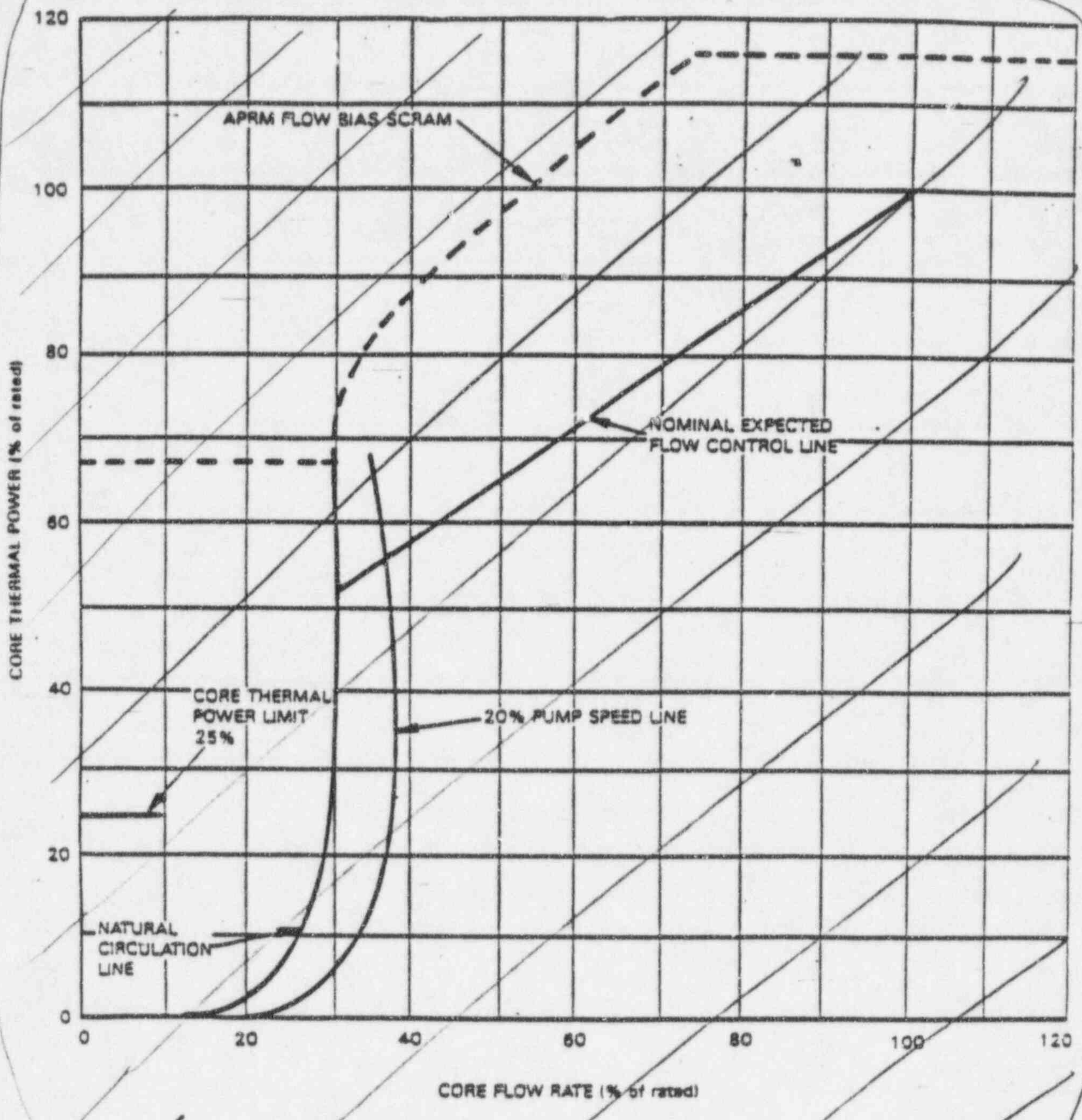


Figure 2.2.1-1. APRM Flow Bias Scram Relationship to Normal Operating Conditions

ITS 3.3.1.1 and Table 3.3.1.1-1

Changed per 96TSB02, ITS

TABLE 4.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[(a)]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	D D	S/U[(b)(c)], W[(d)] W	R R	2 3, 4, 5
b. Inoperative	NA	W[(d)]	NA	2, 3, 4, 5
2. Average Power Range Monitor:				
a. Neutron Flux - High 15%	S S	S/U[(b)(m)], W[(d)] W[(n)]	Q Q	2 5
b. Flow-Biased Simulated Thermal Power - High	S	S/U[(b)], Q	W[(e)](f), Q	1
c. Fixed Neutron Flux - High, 120%	S	S/U[(b)], Q	W[(e)], Q	1
d. Inoperative	NA	Q[(m)(n)]	NA	1, 2, 5
e. Downscale	NA	Q	NA	1
f. LPRM	D	NA	(g)	1, 2, 5
3. Reactor Vessel Steam Dome Pressure - High				
Transmitter:	NA[(k)]	NA	R[(l)]	1, 2
Trip Logic:	D	Q	Q	1, 2
4. Reactor Vessel Water Level - Low, Level 1				
Transmitter:	NA[(k)]	NA	R[(l)]	1, 2
Trip Logic:	D	Q	Q	1, 2

Add SR 3.3.1.1.18

ADD Note 3 to SR 3.3.1.1.11

Changed per 96TSB02, ITS

Changed per 96TSB02, ITS

TABLE 4.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES

(a) Neutron detectors ~~may~~ be excluded from CHANNEL CALIBRATION.

(b) Within 24 hours prior to startup, if not performed within the previous 7 days.

(c) The IRM channels shall be compared to the APRM channels and the SRM instruments for overlap during each startup, if not performed within the previous 7 days.

(d) When changing from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, perform the required surveillance within 12 hours after entering OPERATIONAL CONDITION 2, if not performed within the previous 7 days.

(e) This calibration shall consist of the adjustment of the APRM readout to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

(f) ~~This calibration shall consist of the adjustment of the APRM flow-biased simulated thermal power channel to conform to a calibrated flow signal.~~

(g) The LPRMs shall be calibrated at least once per effective full power month (EFPM) using the TIP system.

(h) This calibration shall consist of a physical inspection and actuation of these position switches.

(i) (Deleted)

SIC 5.10

(j) (Deleted)

(k) The transmitter channel check is satisfied by the trip unit channel check. A separate transmitter check is not required.

(l) ansmitters are exempted from the quarterly channel calibration.

(m) Placement of Reactor Mode Switch into the Startup/Hot Standby position is permitted for the purpose of performing the required surveillance prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

(n) Placement of Reactor Mode Switch into the Shutdown or Refuel position is permitted for the purpose of performing the required surveillance provided all control rods are fully inserted and the vessel head bolts are tensioned.

(o) Surveillance is not required when THERMAL POWER is less than 30% of RATED THERMAL POWER.

Changed per 96TSB02, ITS

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

Changed per 96TSB02, ITS

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant recirculation loops shall be in operation with the cross-tie valve closed, the pump discharge valves OPERABLE, the pump discharge bypass valves OPERABLE or closed and

a. Total core flow shall be greater than or equal to 35 million lbs/hr, or

b. THERMAL POWER shall be less than or equal to the limit specified in Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

Changed per 96TSB02, ITS

ACTION:

Changed per 96TSB02, ITS

a. With both reactor coolant system recirculation loops not in operation, immediately initiate an orderly reduction of THERMAL POWER so that it is less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours, and restore both loops to operation within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. With one reactor coolant system recirculation loop not in operation, immediately initiate either an orderly reduction of THERMAL POWER so that it is less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours or increase core flow so that it is greater than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours, and restore both loops to operation within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c. With two reactor coolant system recirculation loops in operation and total core flow less than 35 million lbs/hr and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1:

   1. Immediately initiate action to reduce THERMAL POWER so that it is less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours, or

   2. Immediately initiate action to increase core flow so that it is greater than 35 million lbs/hr within 2 hours, or

   3. Determine the APRM and LPRM neutron flux noise levels within 2 hours, and:

      a) If the APRM and LPRM neutron flux noise levels are less than three times their established baseline levels or less than 5% peak-to-peak, continue to determine the noise levels at least once per 24 hours and within 1 hour after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER, or

      b) If the APRM or LPRM neutron flux noise levels are greater than or equal to three times their established baseline levels and greater than 5% peak-to-peak, immediately initiate corrective action and restore the noise levels to within the required limits within 2 hours by increasing core flow to greater than 35 million lbs/hr and/or by initiating an orderly reduction of

*See Special Test Exception 3.10.4.

Changed per 96TSB02, ITS

LIMITING CONDITION FOR OPERATION (Continued)

Changed per 96TSB02, ITS

ACTION: (Continued)

THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1.

Changed per 96TSB02, ITS

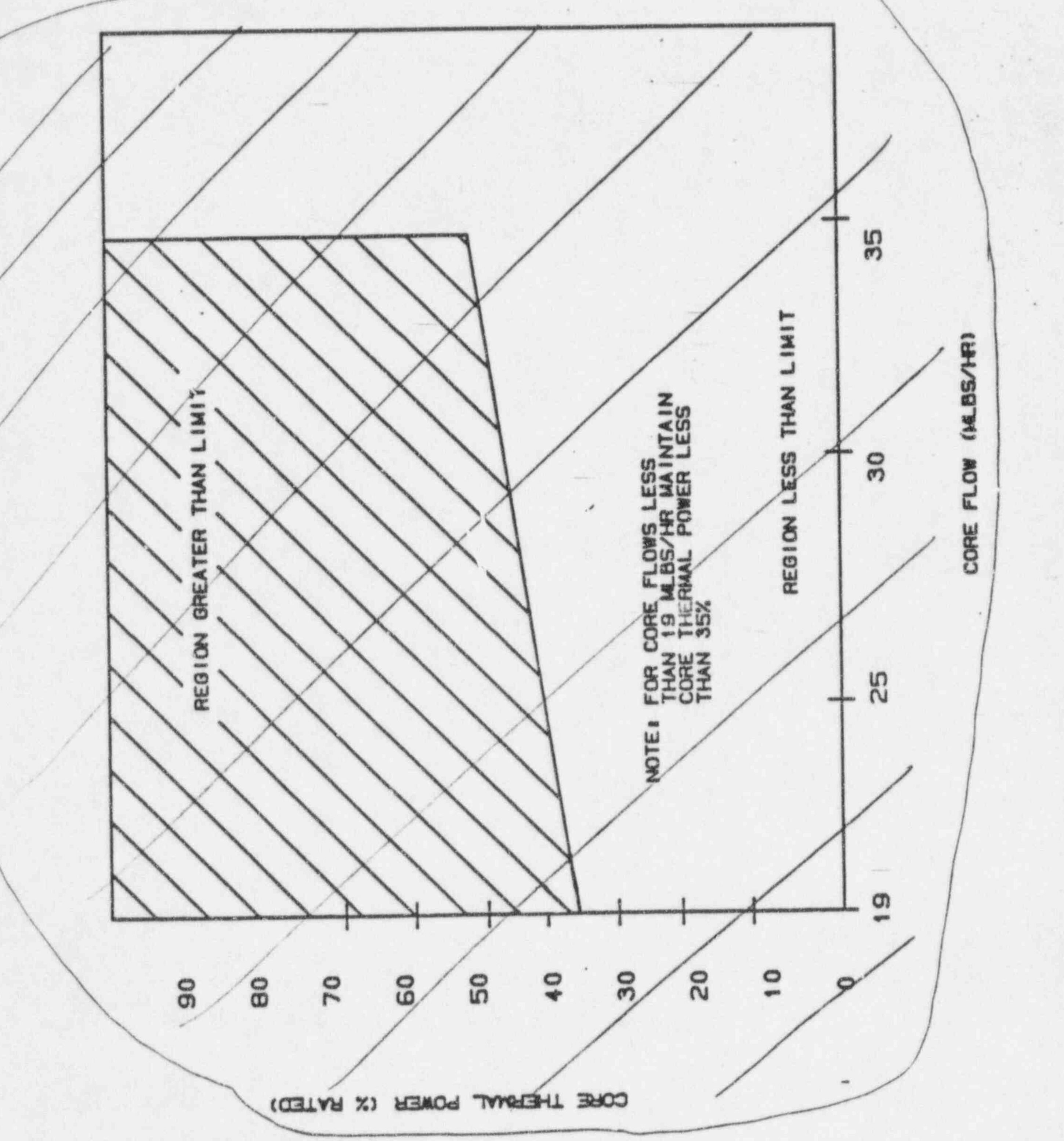
SURVEILLANCE REQUIREMENTS

4.4.1.1.1 Each pump discharge valve and bypass valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each COLD SHUTDOWN which exceeds 48 hours, if not performed in the previous 31 days.

4.4.1.1.2 Each pump discharge bypass valve, if not OPERABLE, shall be verified to be closed at least once per 31 days.

4.4.1.1.3 Establish baseline APRM and LPRM neutron flux noise values at a point below the 100% rated rod line during startup testing following each refueling outage.

FIGURE 4.1.1-1
THERMAL POWER LIMITATIONS



ITS 5.6.5

5.6.5.a.3 The Allowable Value for Function 2.b, APRM Flow Biased Simulated Thermal Power - High, for Specification 3.3.1.1; and

CORE OPERATING LIMITS REPORT (Continued)

Changed per 96TSB02, ITS

b. The core flow and core power adjustments for Specification 3.2.2.1.

c. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specifications 3.2.2.1 and 3.2.2.2.

d. The rod block monitor upscale trip setpoint and allowable value for Specification 3.3.4.

and shall be documented in the CORE OPERATING LIMITS REPORT.

6.9.3.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

a. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).

Changed per 96TSB02, ITS

Insert 5.6.5.b

b. The May 18, 1984 and October 22, 1984 NRC Safety Evaluation Reports for the Brunswick Reload Methodologies described in:

1. Topical Report NF-1583.01, "A Description and Validation of Steady-State Analysis Methods for Boiling Water Reactors," February 1983.

2. Topical Report NF-1583.02, "Methods of RECORD," February 1983.

3. Topical Report NF-1583.03, "Methods of PRESTO-B," February 1983.

4. Topical Report NF-1583.04, "Verification of CP&L Reference BWR Thermal-Hydraulic Methods Using the FIBWR Code," May 1983.

Changed per 96TSB02, ITS

6.9.3.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

6.9.3.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.10 RECORD RETENTION

Facility records shall be retained in accordance with ANSI-N45.2.9-1974.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

Insert 5.6.5.b

2. NEDO-32339-A, "Reactor Stability LongTerm Solution: Enhanced Option I-A," July 1995.

3. NEDC-32339-P Supplement 1, "Reactor Stability Long Term Solution: Enhanced Option I-A ODYSY Computer Code," March 1994 (Approved in NRC Safety Evaluation dated January 4, 1996).

4. NEDO-32339 Supplement 3, "Reactor Stability Long Term Solution: Enhanced Option I-A Flow Mapping Methodology," August 1995 (Approved in NRC Safety Evaluation dated May 28, 1996).

3/4.4 REACTOR COOLANT SYSTEM

Changed per 96TSB02, ITS

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation with a reactor core coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, evaluated, and determined to be acceptable.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does present a hazard in case of a design basis accident by increasing the blowdown area and eliminating the capability of reflooding the core. Thus, the requirement for shutdown of the facility with a jet pump inoperable.

The established characteristics for the criteria of 4.4.1.2 are bands of values that encompass the normal scatter of the data. The scatter in the data can be attributed primarily to monitoring inaccuracies and instrumentation tolerances. An evaluation of these factors will be used to determine the widths of the bands. The bands will be centered about the expected values determined from operating data. The acceptance criteria will be these bands plus the appropriate percentage of the expected values at any point. The acceptance criteria will be updated as required to reflect any changes in the recirculation system that would affect the monitored variables.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures should be within 50°F of each other prior to start-up of an idle loop.

Since the coolant in the bottom of the vessel is at a lower temperature than the water in the upper regions of the core, undue stress on the vessel would result if the temperature difference were greater than 145°F.

~~Neutron flux noise limits are established to ensure early detection of limit cycle neutron flux oscillations. BWR cores typically operate with neutron flux noise caused by random boiling and flow noise. Typical neutron flux noise levels of 1 to 12% of rated power (peak-to-peak) have been reported for the range of low to high recirculation loop flow during both single and dual recirculation loop operation. Neutron flux noise levels significantly larger than these values are considered in the thermal/mechanical fuel design and are found to be of negligible consequence. In addition, stability tests at operating BWR's have demonstrated that when stability related neutron flux limit cycle oscillations occur they result in peak-to-peak neutron flux limit cycles 5 to 10 times the typical values. Therefore, actions taken to reduce neutron flux noise levels exceeding three (3) times the typical value are sufficient to ensure early detection of limit cycle neutron flux oscillations.~~

~~Data to establish baseline APRM and LPRM neutron flux noise values is obtained at a point below the 100% rated rod line. A minimum of two detectors of one LPRM string per core octant and two detectors of one LPRM string near the center of the core should be monitored. Detectors used for monitoring should be selected to provide core wide representation. Substitutions are permitted for inoperable LPRM detectors.~~

Changed per 96TSB02, ITS

These specifications are based on the guidance of General Electric SIL #380, Rev. 1, 2-10-84.

3/4.4.2 SAFETY/RELIEF VALVES

The reactor coolant system safety valve function of the safety-relief valves operate to prevent the system from being pressurized above the Safety Limit of 1325 psig. The system is designed to meet the requirements of the ASME Boiler and Pressure Vessel Code Section III for the pressure vessel and ANSI B31.1, 1967, Code for the reactor coolant system piping.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE
3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shut down to allow further investigation and corrective action. Monitoring leakage at eight hour intervals is in conformance with the 12/21/89 NRC SER for GL 88-01.

3/4.4.4 CHEMISTRY

The reactor water chemistry limits are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low; thus, the higher limit on chlorides is permitted during full power operation. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides, and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity outside the limits, additional samples must be examined to ensure that the chlorides are not exceeding the limits.

Changed per 96TSB02, ITS

Insert New Specification 3.2.3 and B 3.2.3

Insert New Specification 3.2.3, "Fraction of Core Boiling Boundary (FCBB)," and associated Bases as shown in Enclosures 6 and 7 for BNP Units 1 and 2, respectively.

Insert New Specification 3.3.1.3 and B 3.3.1.3

Insert New Specification 3.3.1.3, "Period Based Detection System (PBDS)," and associated Bases as shown in Enclosures 6 and 7 for BNP Units 1 and 2, respectively.

ENCLOSURE 6

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
NRC DOCKET NOS. 50-325 AND 50-324
OPERATING LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
ENHANCED OPTION I-A STABILITY TECHNICAL SPECIFICATIONS

<u>TYPED TECHNICAL SPECIFICATION AND BASES PAGES - UNIT 1</u>

3.2 POWER DISTRIBUTION LIMITS

*3.2.3 Fraction of Core Boiling Boundary (FCBB)

LCO 3.2.3 The FCBB shall be ≤ 1.0.

APPLICABILITY: THERMAL POWER and core flow in the Restricted Region as specified in the COLR,
MODE 1 when RPS Function 2.b, APRM Flow Biased Simulated Thermal Power-High, Allowable Value is "Setup" as specified in the COLR.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. FCBB not within limit for reasons other than an unexpected loss of feedwater heating or unexpected reduction in core flow.	A.1	Restore FCBB to within limit.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR ---------NOTE-------- Required Action B.1 and Required Action B.2 shall be completed if this Condition is entered due to an unexpected loss of feedwater heating or unexpected reduction in core flow. --------------------- FCBB not within limit due to an unexpected loss of feedwater heating or unexpected reduction in core flow.	B.1 Initiate action to exit the Restricted Region. AND B.2 Initiate action to return APRM Flow Biased Simulated Thermal Power—High Allowable Value to "non-Setup" value.	Immediately Immediately following exit of Restricted Region

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	-------------------NOTE------------------- Not required to be performed until 15 minutes after entry into the Restricted Region if entry was the result of an unexpected transient. -- Verify FCBB ≤ 1.0.	24 hours <u>AND</u> Once within 15 minutes following unexpected transient

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.8	Calibrate the local power range monitors.	1100 MWD/T average core exposure
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.10	Calibrate the trip units.	92 days
SR 3.3.1.1.11	-------------------NOTES------------------- 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Function 2.b, the digital components of the flow control trip reference cards are excluded. --- Perform CHANNEL CALIBRATION.	92 days
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.13	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	Verify the APRM Flow Biased Simulated Thermal Power—High time constant is ≤ 7 seconds.	24 months
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.16	Verify Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ 30% RTP.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.17	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Functions 3 and 4, the sensor response time may be assumed to be the design sensor response time. 3. For Function 5, "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. -- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS
SR 3.3.1.1.18	Adjust the flow control trip reference card to conform to reactor flow.	Once within 7 days after reaching equilibrium conditions following refueling outage

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux − High	2	3	G	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.2.2.15	NA
2. Average Power Range Monitors					
a. Neutron Flux − High, Startup	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 15% RTP
b. Flow Biased Simulated Thermal Power − High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.18	(b)

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Allowable Values specified in the COLR.

3.3 INSTRUMENTATION

3.3.1.3 Period Based Detection System (PBDS)

LCO 3.3.1.3 One channel of PBDS instrumentation shall be OPERABLE.

<u>AND</u>

Each OPERABLE channel of PBDS instrumentation shall not indicate High—High Alarm.

APPLICABILITY: THERMAL POWER and core flow in the Restricted Region specified in the COLR,
THERMAL POWER and core flow in the Monitored Region specified in the COLR.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Any OPERABLE PBDS channel indicating High—High Alarm.	A.1	Manually scram the reactor.	Immediately
B. Required PBDS channel inoperable while in the Restricted Region.	B.1	--------NOTE-------- Only applicable if RPS Function 2.b, APRM Flow Biased Simulated Thermal Power—High, Allowable Value is "Setup". --------------------	
		Initiate action to exit the Restricted Region.	Immediately
	<u>OR</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Manually scram the reactor.	Immediately
C. Required PBDS channel inoperable while in the Monitored Region.	C.1 Initiate action to exit the Monitored Region.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.3.1	Verify each OPERABLE channel of PBDS instrumentation not in High-High Alarm.	12 hours
SR 3.3.1.3.2	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.3.3	Perform CHANNEL FUNCTIONAL TEST.	24 months

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1;

2. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.2;

3. The Allowable Value for Function 2.b, APRM Flow Biased Simulated Thermal Power—High, for Specification 3.3.1.1; and

4. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).

2. NEDO-32339-A, "Reactor Stability Long Term Solution: Enhanced Option I-A," July 1995.

3. NEDC-32339-P Supplement 1, "Reactor Stability Long Term Solution: Enhanced Option I-A ODYSY Computer Code," March 1994 (Approved in NRC Safety Evaluation dated January 4, 1996).

4. NEDO-32339 Supplement 3, "Reactor Stability Long Term Solution: Enhanced Option I-A Flow Mapping Methodology," August 1995 (Approved in NRC Safety Evaluation dated May 28, 1996).

5. NRC Safety Evaluation for Brunswick Unit 1 Amendment No. 182.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 Fraction of Core Boiling Boundary (FCBB)

BASES

BACKGROUND

General Design Criteria 12 requires protection of fuel thermal safety limits from conditions caused by neutronic/thermal hydraulic instability. Neutronic/thermal hydraulic instability result in power oscillations which could result in exceeding the MCPR Safety Limit (SL). The MCPR SL is set such that 99.9% of the fuel rods avoid boiling transition during normal operation and during an anticipated operational occurrence (AOO) (refer to the Bases for SL 2.1.1.2).

The FCBB is the ratio of the power generated in the lower 4 feet of the active reactor core to the power required to produce bulk saturated boiling of the coolant entering the fuel channels. The value of 4 feet above the bottom of the active fuel is set as the boiling boundary limit based on analysis described in Section 9 of Reference 1. The boiling boundary limit is established to ensure that the core will remain stable during normal reactor operations in the Restricted Region of the power and flow map defined in the COLR which may otherwise be susceptible to neutronic/thermal hydraulic instabilities.

Planned operation in the Restricted Region is accommodated by manually establishing the "Setup" Allowable Values for the APRM Flow-Biased Simulated Thermal Power—High scram and control rod block functions. The "Setup" Allowable Values of the APRM Flow-Biased Thermal Power—High Function (refer to LCO 3.3.1.1, Table 3.3.1.1-1, Function 2.b.) are consistent with assumed operation in the Restricted Region with FCBB $\leq$ 1.0. Operation with the "Setup" values enables entry into the Restricted Region without a control rod block that would otherwise occur. Plant operation with the "Setup" values is limited as much as practical due to the effects on plant operation required to meet the FCBB limit.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in establishing the boiling boundary limit are presented in Section 9 of Reference 1. Operation with the FCBB $\leq$ 1.0 (i.e., a bulk saturated boiling boundary $\geq$ 4 feet) is expected to ensure that operation within the Restricted Region will not result

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

in neutronic/thermal hydraulic instability due to either steady-state operation or as the result of an AOO which initiates and terminates entirely within the Restricted Region. Analysis also confirms that AOOs initiated from outside the Restricted Region (i.e., without an initial restriction on FCBB) which terminate in the Restricted Region are not expected to result in instability. The types of transients specifically evaluated are loss of flow and coolant temperature decrease which are limiting for the onset of instability (Ref. 1).

Although the onset of instability does not necessarily occur if the FCBB is greater than 1.0 in the Restricted Region, bulk saturated boiling at the 4 foot boiling boundary limit has been adopted so as to preclude neutronic/thermal hydraulic instability during operation in the Restricted Region. The effectiveness of this limit is based on the demonstration (Ref. 1) that with the limit met large margin to the onset of neutronic/thermal hydraulic instability exists and all major state parameters that affect stability have relatively small impacts on stability performance.

The FCBB satisfies Criterion 2 of Reference 2.

LCO

Requiring FCBB ≤ 1.0 ensures the bulk coolant boiling boundary is ≥ 4 feet from the bottom of the active core. Analysis (Ref. 1) has shown that for anticipated operating conditions of core power, core flow, axial and radial power shapes, and inlet enthalpy, a boiling boundary of 4 feet ensures variations in these key parameters do not have a significant impact on stability performance.

Neutronic/thermal hydraulic instabilities can result in power oscillations which could result in exceeding the MCPR Safety Limit (SL). The MCPR SL is set such that 99.9% of the fuel rods avoid boiling transition during normal operation and during an AOO (refer to the Bases for SL 2.1.1.2).

APPLICABILITY

The FCBB limit is used to prevent core conditions necessary for the onset of instability and thereby preclude neutronic/thermal hydraulic instability while operating in the Restricted Region defined in the COLR.

(continued)

BASES

APPLICABILITY (continued)

The boundary of the Restricted Region in the Applicability of this LCO is analytically established in terms of thermal power and core flow. The Restricted Region is defined by the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints, which are a function of reactor recirculation drive flow. The Restricted Region Entry Alarm (RREA) signal is generated by the Flow Control Trip Reference (FCTR) card using the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints. As a result, the RREA is coincident with the Restricted Region boundary under all anticipated operating conditions when the setpoints are not "Setup," and provides indication of entry into the Restricted Region. However, APRM Flow Biased Simulated Thermal Power—High Control Rod Block signals provided by the FCTR card, that are not coincident with the Restricted Region boundary, do not generate a valid RREA. The Restricted Region boundary for this LCO Applicability is specified in the COLR.

When the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints are "Setup," the applicable setpoints used to generate the RREA are moved to the interior boundary of the Restricted Region to allow controlled operation within the Restricted Region. While the setpoints are "Setup," the Restricted Region boundary remains defined by the normal APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints. Parameters, such as reactor power and core flow available at the reactor controls, may be used to provide immediate confirmation that entry into the Restricted Region could reasonably have occurred.

The FCBB limit is also used to ensure that core conditions, while operating with "Setup" values, remain consistent with analyzed transients initiated from inside and outside the Restricted Region.

Operation outside the Restricted Region is not susceptible to neutronic/thermal hydraulic instability when applicable thermal power distribution limits such as MCPR are met.

ACTIONS

A.1

If FCBB is not within the required limit, core conditions necessary for the onset of neutronic/hydraulic thermal instability may result. Therefore, prompt action is taken

(continued)

ACTIONS

A.1 (continued)

to restore the FCBB to within the limit such that the stability of the core can be assured. Following uncontrolled entry into the Restricted Region, prompt restoration of FCBB within limit can be expected if FCBB is known to not significantly exceed the limit. Therefore, efforts to restore FCBB within limit following an uncontrolled entry into the Restricted Region are appropriate if operation prior to entry was consistent with planned entry or the potential for entry was recognized as demonstrated by FCBB being monitored and known to not significantly exceed the limit. Actions to exit the Restricted Region are appropriate when FCBB can not be expected to be restored in a prompt manner.

Actions to restart an idle recirculation loop, withdraw control rods or reduce recirculation flow may result in approaching unstable reactor conditions and are not allowed to be used to comply with this Required Action. The 2 hour Completion Time is based on engineering judgement as to a reasonable time to restore the FCBB to within limit. The 2 hour Completion Time is acceptable based on the availability of the PBDS per Specification 3.3.1.3, "Period Based Detection System" and the low probability of a neutronic/thermal hydraulic instability event.

B.1 and B.2

Changes in reactor core state conditions resulting from an unexpected loss of feedwater heating or unexpected reduction in core flow (e.g., any unexpected reduction in feedwater temperature, recirculation pump trip, or recirculation pump run back) require immediate initiation of action to exit the Restricted Region and return the APRM Flow Biased Simulated Thermal Power—High Function (refer to LCO 3.3.1.1, Table 3.3.1.1-1, Function 2.b.) to the "non-Setup" value. Condition B is modified by a Note that specifies that Required Actions B.1 and B.2 must be completed if this Condition is entered due to an unexpected loss of feedwater heating or unexpected reduction in core flow. The completion of Required Actions B.1 and B.2 is required even though FCBB may be calculated and determined to be within limit. Core conditions continue to change after an unexpected loss of feedwater heating or unexpected reduction in core flow due to transient induced changes with the

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

potential that the FCBB may change and the limit not be met. The potential for changing core conditions, with FCBB not met, is not consistent with operation in the Restricted Region or with the APRM Flow Biased Simulated Thermal Power—High Function "Setup". Therefore, actions to exit the Restricted Region and return the APRM Flow Biased Simulated Thermal Power—High Function to the "non-Setup" value are required to be completed in the event Condition B is entered due to an unexpected loss of feedwater heating or an unexpected reduction in core flow.

If Operator actions to restore the FCBB to within limit are not successful within the specified Completion Time of Condition A, reactor operating conditions may be changing and may continue to change such that core conditions necessary for the onset of neutronic/thermal hydraulic instability may be met. Therefore, in the event the Required Action and associated Completion Time of Condition A is not met, immediate action to exit the Restricted Region and return the APRM Flow Biased Simulated Thermal Power—High Function to the "non-Setup" value is required.

Exit of the Restricted Region can be accomplished by control rod insertion and/or recirculation flow increases. Actions to restart an idle recirculation loop, withdraw control rods or reduce recirculation flow may result in approaching unstable reactor conditions and are not allowed to be used to comply with this Required Action. The time required to exit the Restricted Region will depend on existing plant conditions. Provided efforts are begun without delay and continued until the Restricted Region is exited, operation is acceptable.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

Verifying FCBB ≤ 1.0 is required to ensure the reactor is operating within the assumptions of the safety analysis. The boiling boundary limit is established to ensure that the core will remain stable during normal reactor operations in the Restricted Region of the power and flow map defined in the COLR which may otherwise be susceptible to neutronic/thermal hydraulic instabilities.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1 (continued)

FCBB is required to be verified every 24 hours while operating in the Restricted Region defined in the COLR. The 24 hour Frequency is based on both engineering judgment and recognition of the slow rate of change in power distribution during normal operation.

The second Frequency requires FCBB to be within the limit within 15 minutes following an unexpected transient. The verification of the FCBB is required as a result of the possibility that the unexpected transient results in the limit not being met. The 15 minute frequency is based on both engineering judgement and the availability of the PBDS to provide the operator with information regarding the potential imminent onset of neutronic/thermal hydraulic instability. The 15 minute Frequency for this SR is not to be used to delay entry into Condition B following an unexpected reduction in feedwater heating, recirculation pump trip, or recirculation pump run back.

This Surveillance is modified by a Note which allows 15 minutes to verify FCBB following entry into the Restricted Region if the entry was the result of an unexpected transient (i.e., an unintentional or unplanned change in core thermal power or core flow). The 15 minute allowance is based on both engineering judgement and the availability of the PBDS to provide the operator with information regarding the potential imminent onset of neutronic/thermal hydraulic instability. The 15 minute allowance of the Note is not to be used to delay entry into Condition B if the entry into the Restricted Region was the result of an unexpected reduction in feedwater heating, recirculation pump trip, or recirculation pump run back.

REFERENCES

1. NEDO 32339-A, Reactor Stability Long Term Solution: Enhanced Option I-A, July 1995.

2. 10 CFR 50.36(c)(2)(ii).

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux—High, Startup (continued)

this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 11 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux—High, Startup Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

In MODE 1, the Average Power Range Monitor Flow Biased Simulated Thermal Power—High and Fixed Neutron Flux—High Functions provide protection against reactivity transients and the RWM and Rod Block Monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power—High

The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of rated recirculation drive flow (W) in percent and is clamped at an upper limit that is always lower than the Average Power Range Monitor Fixed Neutron Flux—High Function Allowable Value. The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function provides a general definition of the licensed core power/core flow operating domain.

The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function is not associated with an LSSS. Operating limits established for the licensed operating domain are used to develop the Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function Allowable Values to provide pre-emptive reactor scram and prevent

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power—High (continued)

gross violation of the licensed operating domain. Operation outside the licensed operating domain may result in anticipated operational occurrences and postulated accidents being initiated from conditions beyond those assumed in the safety analysis. Operation within the licensed operating domain also ensures compliance with General Design Criterion 12.

General Design Criterion 12 requires protection of fuel thermal safety limits from conditions caused by neutronic/thermal hydraulic instability. Neutronic/thermal hydraulic instabilities result in power oscillations which could result in exceeding the MCPR SL.

The area of the core power and flow operating domain susceptible to neutronic/thermal hydraulic instability is affected by the Fraction of Core Boiling Boundary (LCO 3.2.3, FCBB). "Setup" and normal ("non-Setup") Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function Allowable Values are specified in the Core Operating Limits Report (COLR). When the normal value is selected protection against neutronic/thermal hydraulic instability is provided by preventing operation in the susceptible area of the operating domain during operation outside the Restricted Region of the operating domain specified in the COLR with the FCBB limit not required to be met. When the "Setup" value is selected meeting the FCBB limit provides protection against neutronic/thermal hydraulic instability.

"Setup" and "non-Setup" values are selected by operator manipulation of the recessed Setup button on each flow control trip reference card. Selection of the "Setup" value is intended only for planned operation in the Restricted Region as specified in the COLR. Operation in the Restricted Region with the Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function in "Setup" requires the FCBB limit to be met and is not generally consistent with normal power operation.

The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function uses a trip level generated by the flow control trip reference card based on recirculation loop drive flow. The proper representation of drive flow as

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power—High (continued)

indication of core flow is ensured through drive flow alignment. This is accomplished by the selection of appropriate dip switch positions on the flow control trip reference cards (Refer to SR 3.3.1.1.18). Changes in the core flow to drive flow functional relationship may vary over the core flow operating range. These changes can result from gradual changes in the Recirculation System and core components over the reactor life time as well as specific maintenance performed on these components (e.g., jet pump cleaning).

The APRM System is divided into two groups of channels with three APRM inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Flow Biased Simulated Thermal Power—High with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 11 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located. Each APRM channel receives two total drive flow signals representative of total core flow. The total drive flow signals are generated by four flow units, two of which supply signals to the trip system A APRMs, while the other two supply signals to the trip system B APRMs. Each flow unit signal is provided by summing up the flow signals from the two recirculation loops. To obtain the most conservative reference signals, the total flow signals from the two flow units (associated with a trip system as described above) are routed to a low auction circuit associated with each APRM. Each APRM's auction circuit selects the lower of the two flow unit signals for use as the scram trip reference for that particular APRM. Each required Average Power Range Monitor Flow Biased Simulated Thermal Power—High channel only requires an input from one OPERABLE flow unit, since the individual APRM channel will perform the intended function with only one OPERABLE flow unit input. However, in order to maintain single failure criteria as described above for the Function, at least one required Average Power Range

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power—High (continued)

Monitor Flow Biased Simulated Thermal Power—High channel in each trip system must be capable of maintaining an OPERABLE flow unit signal in the event of a failure of an auction circuit, or a flow unit, in the associated trip system (e.g., if a flow unit is inoperable, one of the two required Average Power Range Monitor Flow Biased Simulated Thermal Power—High channels in the associated trip system must be considered inoperable).

The THERMAL POWER time constant of ≤ 7 seconds is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER.

The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function is required to be OPERABLE in MODE 1 when there is the possibility of neutronic/thermal hydraulic instability. The potential to exceed the SL applicable to high pressure and core flow conditions (MCPR SL), which provides fuel cladding integrity protection, exists if neutronic/thermal hydraulic instability occurs. During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

2.c. Average Power Range Monitor Fixed Neutron Flux—High

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The Average Power Range Monitor Fixed Neutron Flux—High Function is capable of generating a trip signal without the electronically filtered time constant to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of References 4 and 7, the Average Power Range Monitor Fixed Neutron Flux—High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (SRVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 2) takes credit for the Average Power Range Monitor Fixed Neutron Flux—High Function to terminate the CRDA.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.11 and SR 3.3.1.1.13 (continued)

calibration (SR 3.3.1.1.3) and the 1100 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.8). A second Note is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR. Note 3 to SR 3.3.1.1.11 states that the digital components of the flow control trip reference card are excluded from CHANNEL CALIBRATION of Function 2.b, Average Power Range Monitor Flow Biased Simulated Thermal Power—High. The analog output potentiometers of the flow control trip reference card are not excluded from this test. The flow control trip reference card has an automatic self-test feature which periodically tests the hardware that performs the digital algorithm. Exclusion of the digital components of the flow control trip reference card from CHANNEL CALIBRATION of Function 2.b is based on conditions required to perform the test and the small likelihood of a change in the status of these components not being detected.

The Frequency of SR 3.3.1.1.11 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.13 is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.1.14

The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function uses an electronic filter circuit to generate a signal proportional to the core THERMAL POWER from the APRM neutron flux signal. This filter circuit is representative of the fuel heat transfer dynamics that produce the relationship between the neutron flux and the core THERMAL POWER. The filter time constant must be verified to be ≤ 7 seconds to ensure that the channel is accurately reflecting the desired parameter.

The Frequency of 24 months is based on engineering judgment considering the reliability of the components.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.17

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. This test may be performed in one measurement or in overlapping segments, with verification that all components are tested. The RPS RESPONSE TIME acceptance criteria are included in Reference 13.

As noted (Note 1), neutron detectors for Function 2 are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time. In addition, Note 2 states the response time of the sensors for Functions 3 and 4 may be assumed in the RPS RESPONSE TIME test to be the design sensor response time. This is allowed since the sensor response time is a small part of the overall RPS RESPONSE TIME (Ref. 14).

RPS RESPONSE TIME tests are conducted on a 24 month STAGGERED TEST BASIS. Note 3 requires STAGGERED TEST BASIS Frequency to be determined based on 4 channels per trip system, in lieu of the 8 channels specified in Table 3.3.1.1-1 for the MSIV Closure Function. This Frequency is based on the logic interrelationships of the various channels required to produce an RPS scram signal. The 24 month Frequency is consistent with the BNP refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

SR 3.3.1.1.18

The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function uses a trip level generated by the flow control trip reference card based on the recirculation loop drive flow. The drive flow is adjusted by a digital algorithm according to selected drive flow alignment dip switch settings. This SR sets, as necessary, the flow control trip reference card to ensure the drive flow alignment used results in the appropriate trip level being generated from the digital components of the card.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.18 (continued)

The Frequency of once within 7 days after reaching equilibrium conditions following a refueling outage is based on the expectation that any change in the core flow to drive flow functional relationship during power operation would be gradual and the maintenance on the Recirculation System and core components which may impact the relationship is expected to be performed during refueling outages. The 7 day time period to reach equilibrium conditions is based on plant conditions required to perform the test, engineering judgment of the time required to collect and analyze the necessary flow data, and engineering judgment of the time required to adjust and check the adjustment of each flow control trip reference card. The 7 day time period to reach equilibrium conditions is acceptable based on the low probability of a neutronic/thermal hydraulic instability event.

REFERENCES

1. UFSAR, Section 7.2.

2. UFSAR, Chapter 15.0.

3. UFSAR, Section 7.2.2.

4. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and ^ September 1995.

5. 10 CFR 50.36(c)(2)(ii).

6. NEDO-23842, Continuous Control Rod Withdrawal in the Startup Range, April 18, 1978.

7. UFSAR, Section 5.2.2.

8. UFSAR, Appendix 5.2A.

9. UFSAR, Section 6.3.1.

10. P. Check (NRC) letter to G. Lainas (NRC), BWR Scram Discharge System Safety Evaluation, December 1, 1980.

11. NEDC-30851-P-A, Technical Specification Improvement Analyses for BWR Reactor Protection System, March 1988.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.1.3 Period Based Detection System (PBDS)

BASES

BACKGROUND

General Design Criteria 12 requires protection of fuel thermal safety limits from conditions caused by neutronic/thermal hydraulic instability. Neutronic/thermal hydraulic instabilities can result in power oscillations which could result in exceeding the MCPR Safety Limit (SL). The MCPR SL ensures that at least 99.9% of the fuel rods avoid boiling transition during normal operation and during an anticipated operational occurrence (AOO) (refer to the Bases for SL 2.1.1.2).

The PBDS provides the operator with an indication that conditions consistent with a significant degradation in the stability performance of the reactor core has occurred and the potential for imminent onset of neutronic/thermal hydraulic instability may exist. Indication of such degradation is cause for the operator to initiate an immediate reactor scram if the reactor is being operated in either the Restricted Region or Monitored Region. The Restricted Region and Monitored Region are defined in the COLR.

The PBDS instrumentation of the Neutron Monitoring System (NMS) consists of two channels. PBDS channel A includes input from 13 local power range monitors (LPRMs) within the reactor core and PBDS channel B includes input from 11 LPRMs within the reactor core. All LPRMs are utilized from each of the axial levels except for the D level detectors. These inputs are continually monitored by the PBDS for variations in the neutron flux consistent with the onset of neutronic/thermal hydraulic instability. Each channel includes separate local indication and separate control room High—High Alarms. While, this LCO specifies OPERABILITY requirements only for one monitoring and indication channel of the PBDS, if both are OPERABLE, a High—High Alarm from either channel results in the need for the operator to take actions.

The primary PBDS component is a card in the NMS with analog inputs and digital processing. The PBDS card has an automatic self-test feature to periodically test the hardware circuit. The self-test functions are executed during their allocated portion of the executive loop

(continued)

BASES

BACKGROUND (continued)

sequence. Any self-test failure indicating loss of critical function results in a common control room "Inoperative" alarm. The inoperable condition is also displayed by an indicating light on the card front panel. A manually initiated internal test sequence can be actuated via a recessed push button. This internal test consists of simulating alarm and inoperable conditions to verify card OPERABILITY. Further descriptions of the PBDS are provided in References 1 and 2.

Actuation of the PBDS High—High Alarm is not postulated to occur due to neutronic/thermal hydraulic instability during operation outside the Restricted Region and the Monitored Region. Periodic perturbations can be introduced into the thermal hydraulic behavior of the reactor core from external sources such as recirculation system components and the pressure and feedwater control systems. These perturbations can potentially drive the neutron flux to oscillate within a frequency range expected for neutronic/thermal hydraulic instability. The presence of such oscillations may be recognized by the period based algorithm of the PBDS and could result in a High—High Alarm. Actuation of the PBDS High—High Alarm outside the Restricted Region and the Monitored Region indicate the presence of a source external to the reactor core and are not indications of neutronic/thermal hydraulic instability.

APPLICABLE SAFETY ANALYSES

Analysis, as described in Section 4 of Reference 1, confirms that AOOs initiated from outside the Restricted Region without stability control and from within the Restricted Region with stability control are not expected to result in neutronic/thermal hydraulic instability. The stability control applied in the Restricted Region (refer to LCO 3.2.3, "Fraction of Core Boiling Boundary (FCBB)") is established to prevent neutronic/thermal hydraulic instability during operation in the Restricted Region. Operation in the Monitored Region is only susceptible to instability under operating conditions beyond those analyzed in Reference 1. The types of transients specifically evaluated are loss of flow and coolant temperature decrease which are limiting for the onset of instability.

The initial conditions assumed in the analysis are reasonably conservative and the immediate post-event reactor conditions are significantly stable. However, these assumed initial conditions do not bound each individual parameter

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

which impacts stability performance (Ref. 1). The PBDS instrumentation provides the operator with an indication that conditions consistent with a significant degradation in the stability performance of the reactor core has occurred and the potential for imminent onset of neutronic/thermal hydraulic instability may exist. Such conditions are only postulated to result from events initiated from initial conditions beyond the conditions assumed in the safety analysis (refer to Section 4, Ref. 1).

The PBDS has no safety function and is not assumed to function during any UFSAR design basis accident or transient analysis. However, the PBDS provides the only indication of the imminent onset of neutronic/thermal hydraulic instability during operation in regions of the operating domain potentially susceptible to instability. Therefore, the PBDS is included in the Technical Specifications.

LCO

One PBDS channel is required to be OPERABLE with a minimum of eight LPRM inputs to monitor reactor neutron flux for indications of imminent onset of neutronic/thermal hydraulic instability. A PBDS channel may be considered OPERABLE with six LPRM inputs when the distribution of OPERABLE LPRMs provides: a) at least one OPERABLE LPRM in each core quadrant or b) at least two OPERABLE LPRMs in the core quadrant opposite any core quadrant with no OPERABLE LPRMs. The required distribution of the LPRMs when a PBDS channel is considered OPERABLE with as few as six OPERABLE LPRMs ensures a minimum of two OPERABLE LPRMs in opposite core quadrants. This distribution ensures that, for all postulated orientations and modes of oscillation, there are at least two OPERABLE LPRMs in the core quadrants in which the local neutron flux will oscillate with a frequency within the range monitored by the PBDS. OPERABILITY requires the ability for the operator to be immediately alerted to a High—High Alarm. This is accomplished by the instrument channel control room alarm. The LCO also requires reactor operation be such that the High—High Alarm is not actuated by any OPERABLE PBDS instrumentation channel.

APPLICABILITY

At least one of two PBDS instrumentation channels is required to be OPERABLE during operation in either the Restricted Region or the Monitored Region specified in the

(continued)

BASES

APPLICABILITY (continued)

COLR. Similarly, operation with the PBDS High—High Alarm of any OPERABLE PBDS instrumentation channel is not allowed in the Restricted Region or the Monitored Region. Operation in these regions is susceptible to instability (refer to the Bases for LCO 3.2.3 and Section 4 of Ref. 1). OPERABILITY of at least one PBDS instrumentation channel and operation with no indication of a PBDS High—High Alarm from any OPERABLE PBDS instrumentation channel is therefore required during operation in these regions.

The boundary of the Restricted Region in the Applicability of this LCO is analytically established in terms of thermal power and core flow. The Restricted Region is defined by the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints, which are a function of reactor recirculation drive flow. The Restricted Region Entry Alarm (RREA) signal is generated by the Flow Control Trip Reference (FCTR) card using the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints. As a result, the RREA is coincident with the Restricted Region boundary under all anticipated operating conditions when the setpoints are not "Setup," and provides the indication regarding entry into the Restricted Region. However, APRM Flow Biased Simulated Thermal Power—High Control Rod Block signals provided by the FCTR card, that are not coincident with the Restricted Region boundary, do not generate a valid RREA. The Restricted Region boundary for this LCO Applicability is specified in the COLR.

When the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints are "Setup" the applicable setpoints used to generate the RREA are moved to the interior boundary of the Restricted Region to allow controlled operation within the Restricted Region. While the setpoints are "Setup" the Restricted Region boundary remains defined by the normal APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints. Parameters such as reactor power and core flow available at the reactor controls, may be used to provide immediate confirmation that entry into the Restricted Region could reasonably have occurred. The Monitored Region in the Applicability of this LCO is analytically established in terms of thermal power and core flow. However, unlike the Restricted Region boundary the Monitored Region is not specifically monitored by plant instrumentation to provide automatic indication of entry into the region. Therefore, the Monitored Region

(continued)

BASES

APPLICABILITY (continued)

boundary is defined solely in terms of thermal power and core flow. The Monitored Region boundary for this LCO Applicability is specified in the COLR.

Operation outside the Restricted Region and the Monitored Region is not susceptible to neutronic/thermal hydraulic instability even under extreme postulated conditions.

ACTIONS

A.1

If at any time while in the Restricted Region or Monitored Region, an OPERABLE PBDS instrumentation channel indicates a High—High Alarm, the operator is required to initiate an immediate reactor scram. Verification that the High—High Alarm is valid may be performed without delay against another output from a PBDS card observable from the reactor controls in the control room prior to the manual reactor scram. This provides assurance that core conditions leading to neutronic/thermal hydraulic instability will be mitigated. This Required Action and associated Completion Time does not allow for evaluation of circumstances leading to the High—High Alarm prior to manual initiation of reactor scram.

B.1 and B.2

Operation with the APRM Flow Biased Simulated Thermal Power—High Function (refer to LCO 3.3.1.1, Table 3.3.1.1-1, Function 2.b) "Setup" requires the stability control applied in the Restricted Region (refer to LCO 3.2.3) to be met. Requirements for operation with the stability control met are established to prevent reactor thermal hydraulic instability during operation in the Restricted Region. When the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints are not "Setup" uncontrolled entry into the Restricted Region is identified by receipt of a valid RREA. Immediate confirmation that the RREA is valid and indicates an actual entry into the Restricted Region may be performed without delay. Immediate confirmation constitutes observation that plant parameters immediately available at the reactor controls (e.g., core power and core flow) are reasonably consistent with entry into the Restricted Region. This immediate confirmation may also constitute recognition that plant parameters are rapidly changing during a

(continued)

ACTIONS

B.1 and B.2 (continued)

transient (e.g., a recirculation pump trip) which could reasonably result in entry into the Restricted Region. While the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints are "Setup," operation in the Restricted Region may be confirmed by use of plant parameters such as reactor power and core flow available at the reactor controls. With the required PBDS channel inoperable while in the Restricted Region, the ability to monitor conditions indicating the potential for imminent onset of neutronic/thermal hydraulic instability as a result of unexpected transients is lost. Therefore, action must be immediately initiated to exit the Restricted Region.

Exit of the Restricted Region can be accomplished by control rod insertion and/or recirculation flow increases. Actions to restart an idle recirculation loop, withdraw control rods or reduce recirculation flow may result in unstable reactor conditions and are not allowed to be used to comply with this Required Action.

The time required to exit the Restricted Region will depend on existing plant conditions. Provided efforts are begun without delay and continued until the Restricted Region is exited, operation is acceptable based on the low probability of a transient which degrades stability performance occurring simultaneously with the required PBDS channel inoperable.

Required Action B.1 is modified by a Note that specifies that initiation of action to exit the Restricted Region only applies if the APRM Flow Biased Simulated Thermal Power—High Function is "Setup". Operation in the Restricted Region without the APRM Flow Biased Simulated Thermal Power—High Function "Setup" indicates uncontrolled entry into the Restricted Region. Uncontrolled entry is consistent with the occurrence of unexpected transients, which, in combination with the absence of stability controls being met may result in significant degradation of stability performance. Under these conditions with the required PBDS instrumentation channel inoperable, the ability to monitor conditions indicating the potential for imminent onset of neutronic/thermal hydraulic instability is lost and continued operation is not justified. Therefore, Required Action B.2 requires immediate reactor scram.

(continued)

BASES

ACTIONS (continued)

C.1

In the Monitored Region the PBDS High—High Alarm provides indication of degraded stability performance. Although not anticipated, operation in the Monitored Region is susceptible to neutronic/thermal hydraulic instability under postulated conditions exceeding those previously assumed in the safety analysis. With the required PBDS channel inoperable while in the Monitored Region, the ability to monitor conditions indicating the potential for imminent onset of neutronic/thermal hydraulic instability is lost. Therefore, action must be initiated to exit the Monitored Region.

Actions to restart an idle recirculation loop, withdraw control rods or reduce recirculation flow may result in approaching unstable reactor conditions and are not allowed to be used to comply with this Required Action. Exit of the Monitored Region is accomplished by control rod insertion and/or recirculation flow increases. However, actions which reduce recirculation flow are allowed provided the FCBB is recently (within 15 minutes) verified to be ≤ 1.0. Recent verification of FCBB being met, provides assurance that with the PBDS inoperable, planned decreases in recirculation drive flow should not result in significant degradation of core stability performance.

The Completion Time of 15 minutes ensures timely operator action to exit the region consistent with the low probability that reactor conditions exceed the initial conditions assumed in the safety analysis. The time required to exit the Monitored Region will depend on existing plant conditions. Provided efforts are begun within 15 minutes and continued until the Monitored Region is exited, operation is acceptable based on the low probability of a transient which degrades stability performance occurring simultaneously with the required PBDS channel inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3.1

During operation in the Restricted Region or the Monitored Region the PBDS High—High Alarm is relied upon to indicate conditions consistent with the onset of neutronic/thermal hydraulic instability. Verification that each OPERABLE

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3.1 (continued)

channel of PBDS instrumentation is not in High—High Alarm every 12 hours provides assurance of the proper indication of the alarm during operation in the Restricted Region or the Monitored Region. The 12 hour Frequency supplements less formal, but more frequent, checks of alarm status during operation.

SR 3.3.1.3.2

Performance of the CHANNEL CHECK every 12 hours ensures that a gross failure of instrumentation has not occurred. This CHANNEL CHECK is normally a comparison of the PBDS indication to the state of the annunciator, as well as comparison to the same parameter on the other channel if it is available. It is based on the assumption that the instrument channel indication agrees with the immediate indication available to the operator, and that instrument channels monitoring the same parameter should read similarly. Deviations between the instrument channels could be an indication of instrument component failure. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL FUNCTIONAL TEST. Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability.

The 12 hour Frequency is based on the CHANNEL CHECK Frequency requirement of similar Neutron Monitoring System components. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.3.3

A CHANNEL FUNCTIONAL TEST is performed for each required PBDS channel to ensure that the system will perform the intended function. The CHANNEL FUNCTIONAL TEST for the PBDS includes manual initiation of an internal test sequence and verification of appropriate alarms and inop conditions being reported.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.3.1.3.3 (continued) Performance of a CHANNEL FUNCTIONAL TEST at a Frequency of 24 months verifies the performance of the PBDS and associated circuitry. The Frequency considers the plant conditions required to perform the test, the ease of performing the test, and the likelihood of a change in the system or component status. The alarm circuit is designed to operate for over 24 months with sufficient accuracy on signal amplitude and signal timing considering environment, initial calibration, and accuracy drift (Ref. 2).
REFERENCES	1. NEDO 32339-A, Reactor Stability Long Term Solution: Enhanced Option I-A, July 1994. 2. NEDC-32339, Supplement 2, Reactor Stability Long Term Solution: Enhanced Option I-A Solution Design. April 1995.

ENCLOSURE 7

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
NRC DOCKET NOS. 50-325 AND 50-324
OPERATING LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
ENHANCED OPTION I-A STABILITY TECHNICAL SPECIFICATIONS

TYPED TECHNICAL SPECIFICATION AND BASES PAGES - UNIT 2

3.2 POWER DISTRIBUTION LIMITS

3.2.3 Fraction of Core Boiling Boundary (FCBB)

LCO 3.2.3 The FCBB shall be $\leq$ 1.0.

APPLICABILITY: THERMAL POWER and core flow in the Restricted Region as specified in the COLR,
MODE 1 when RPS Function 2.b, APRM Flow Biased Simulated Thermal Power-High, Allowable Value is "Setup" as specified in the COLR.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. FCBB not within limit for reasons other than an unexpected loss of feedwater heating or unexpected reduction in core flow.	A.1	Restore FCBB to within limit.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR ---------NOTE-------- Required Action B.1 and Required Action B.2 shall be completed if this Condition is entered due to an unexpected loss of feedwater heating or unexpected reduction in core flow. --------------------- FCBB not within limit due to an unexpected loss of feedwater heating or unexpected reduction in core flow.	B.1	Initiate action to exit the Restricted Region.	Immediately
	AND		
	B.2	Initiate action to return APRM Flow Biased Simulated Thermal Power—High Allowable Value to "non-Setup" value.	Immediately following exit of Restricted Region

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	-------------------NOTE------------------- Not required to be performed until 15 minutes after entry into the Restricted Region if entry was the result of an unexpected transient. -- Verify FCBB ≤ 1.0.	24 hours <u>AND</u> Once within 15 minutes following unexpected transient

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.8	Calibrate the local power range monitors.	1100 MWD/T average core exposure
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.10	Calibrate the trip units.	92 days
SR 3.3.1.1.11	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Function 2.b, the digital components of the flow control trip reference cards are excluded. --- Perform CHANNEL CALIBRATION.	92 days
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.13	------------ -----NOTES----------------- 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	Verify the APRM Flow Biased Simulated Thermal Power—High time constant is ≤ 7 seconds.	24 months
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.16	Verify Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ 30% RTP.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.17	-------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Functions 3 and 4, the sensor response time may be assumed to be the design sensor response time. 3. For Function 5, "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. --- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS
SR 3.3.1.1.18	Adjust the flow control trip reference card to conform to reactor flow.	Once within 7 days after reaching equilibrium conditions following refueling outage

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux — High	2	3	G	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.2.2.15	NA
2. Average Power Range Monitors					
a. Neutron Flux — High, Startup	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 15% RTP
b. Flow Biased Simulated Thermal Power — High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.18	(b)

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Allowable Values specified in the COLR.

3.3 INSTRUMENTATION

3.3.1.3 Period Based Detection System (PBDS)

LCO 3.3.1.3 One channel of PBDS instrumentation shall be OPERABLE.

AND

Each OPERABLE channel of PBDS instrumentation shall not indicate High—High Alarm.

APPLICABILITY: THERMAL POWER and core flow in the Restricted Region specified in the COLR,
THERMAL POWER and core flow in the Monitored Region specified in the COLR.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Any OPERABLE PBDS channel indicating High—High Alarm.	A.1	Manually scram the reactor.	Immediately
B. Required PBDS channel inoperable while in the Restricted Region.	B.1	--------NOTE-------- Only applicable if RPS Function 2.b, APRM Flow Biased Simulated Thermal Power—High, Allowable Value is "Setup". -------------------- Initiate action to exit the Restricted Region.	Immediately
	OR		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Manually scram the reactor.	Immediately
C. Required PBDS channel inoperable while in the Monitored Region.	C.1	Initiate action to exit the Monitored Region.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.3.1	Verify each OPERABLE channel of PBDS instrumentation not in High-High Alarm.	12 hours
SR 3.3.1.3.2	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.3.3	Perform CHANNEL FUNCTIONAL TEST.	24 months

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1;

2. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.2;

3. The Allowable Value for Function 2.b, APRM Flow Biased Simulated Thermal Power—High, for Specification 3.3.1.1; and

4. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).

2. NEDO-32339-A, "Reactor Stability Long Term Solution: Enhanced Option I-A," July 1995.

3. NEDC-32339-P Supplement 1, "Reactor Stability Long Term Solution: Enhanced Option I-A ODYSY Computer Code," March 1994 (Approved in NRC Safety Evaluation dated January 4, 1996).

4. NEDO-32339 Supplement 3, "Reactor Stability Long Term Solution: Enhanced Option I-A Flow Mapping Methodology," August 1995 (Approved in NRC Safety Evaluation dated May 28, 1996).

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 Fraction of Core Boiling Boundary (FCBB)

BASES

BACKGROUND

General Design Criteria 12 requires protection of fuel thermal safety limits from conditions caused by neutronic/thermal hydraulic instability. Neutronic/thermal hydraulic instability result in power oscillations which could result in exceeding the MCPR Safety Limit (SL). The MCPR SL is set such that 99.9% of the fuel rods avoid boiling transition during normal operation and during an anticipated operational occurrence (AOO) (refer to the Bases for SL 2.1.1.2).

The FCBB is the ratio of the power generated in the lower 4 feet of the active reactor core to the power required to produce bulk saturated boiling of the coolant entering the fuel channels. The value of 4 feet above the bottom of the active fuel is set as the boiling boundary limit based on analysis described in Section 9 of Reference 1. The boiling boundary limit is established to ensure that the core will remain stable during normal reactor operations in the Restricted Region of the power and flow map defined in the COLR which may otherwise be susceptible to neutronic/thermal hydraulic instabilities.

Planned operation in the Restricted Region is accommodated by manually establishing the "Setup" Allowable Values for the APRM Flow-Biased Simulated Thermal Power—High scram and control rod block functions. The "Setup" Allowable Values of the APRM Flow-Biased Thermal Power—High Function (refer to LCO 3.3.1.1, Table 3.3.1.1-1, Function 2.b.) are consistent with assumed operation in the Restricted Region with FCBB ≤ 1.0. Operation with the "Setup" values enables entry into the Restricted Region without a control rod block that would otherwise occur. Plant operation with the "Setup" values is limited as much as practical due to the effects on plant operation required to meet the FCBB limit.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in establishing the boiling boundary limit are presented in Section 9 of Reference 1. Operation with the FCBB ≤ 1.0 (i.e., a bulk saturated boiling boundary ≥ 4 feet) is expected to ensure that operation within the Restricted Region will not result

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

in neutronic/thermal hydraulic instability due to either steady-state operation or as the result of an AOO which initiates and terminates entirely within the Restricted Region. Analysis also confirms that AOOs initiated from outside the Restricted Region (i.e., without an initial restriction on FCBB) which terminate in the Restricted Region are not expected to result in instability. The types of transients specifically evaluated are loss of flow and coolant temperature decrease which are limiting for the onset of instability (Ref. 1).

Although the onset of instability does not necessarily occur if the FCBB is greater than 1.0 in the Restricted Region, bulk saturated boiling at the 4 foot boiling boundary limit has been adopted so as to preclude neutronic/thermal hydraulic instability during operation in the Restricted Region. The effectiveness of this limit is based on the demonstration (Ref. 1) that with the limit met large margin to the onset of neutronic/thermal hydraulic instability exists and all major state parameters that affect stability have relatively small impacts on stability performance.

The FCBB satisfies Criterion 2 of Reference 2.

LCO

Requiring FCBB $\leq$ 1.0 ensures the bulk coolant boiling boundary is $\geq$ 4 feet from the bottom of the active core. Analysis (Ref. 1) has shown that for anticipated operating conditions of core power, core flow, axial and radial power shapes, and inlet enthalpy, a boiling boundary of 4 feet ensures variations in these key parameters do not have a significant impact on stability performance.

Neutronic/thermal hydraulic instabilities can result in power oscillations which could result in exceeding the MCPR Safety Limit (SL). The MCPR SL is set such that 99.9% of the fuel rods avoid boiling transition during normal operation and during an AOO (refer to the Bases for SL 2.1.1.2).

APPLICABILITY

The FCBB limit is used to prevent core conditions necessary for the onset of instability and thereby preclude neutronic/thermal hydraulic instability while operating in the Restricted Region defined in the COLR.

(continued)

BASES

APPLICABILITY (continued)

The boundary of the Restricted Region in the Applicability of this LCO is analytically established in terms of thermal power and core flow. The Restricted Region is defined by the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints, which are a function of reactor recirculation drive flow. The Restricted Region Entry Alarm (RREA) signal is generated by the Flow Control Trip Reference (FCTR) card using the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints. As a result, the RREA is coincident with the Restricted Region boundary under all anticipated operating conditions when the setpoints are not "Setup," and provides indication of entry into the Restricted Region. However, APRM Flow Biased Simulated Thermal Power—High Control Rod Block signals provided by the FCTR card, that are not coincident with the Restricted Region boundary, do not generate a valid RREA. The Restricted Region boundary for this LCO Applicability is specified in the COLR.

When the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints are "Setup," the applicable setpoints used to generate the RREA are moved to the interior boundary of the Restricted Region to allow controlled operation within the Restricted Region. While the setpoints are "Setup," the Restricted Region boundary remains defined by the normal APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints. Parameters, such as reactor power and core flow available at the reactor controls, may be used to provide immediate confirmation that entry into the Restricted Region could reasonably have occurred.

The FCBB limit is also used to ensure that core conditions, while operating with "Setup" values, remain consistent with analyzed transients initiated from inside and outside the Restricted Region.

Operation outside the Restricted Region is not susceptible to neutronic/thermal hydraulic instability when applicable thermal power distribution limits such as MCPR are met.

ACTIONS

A.1

If FCBB is not within the required limit, core conditions necessary for the onset of neutronic/hydraulic thermal instability may result. Therefore, prompt action is taken

(continued)

BASES

ACTIONS

A.1 (continued)

to restore the FCBB to within the limit such that the stability of the core can be assured. Following uncontrolled entry into the Restricted Region, prompt restoration of FCBB within limit can be expected if FCBB is known to not significantly exceed the limit. Therefore, efforts to restore FCBB within limit following an uncontrolled entry into the Restricted Region are appropriate if operation prior to entry was consistent with planned entry or the potential for entry was recognized as demonstrated by FCBB being monitored and known to not significantly exceed the limit. Actions to exit the Restricted Region are appropriate when FCBB can not be expected to be restored in a prompt manner.

Actions to restart an idle recirculation loop, withdraw control rods or reduce recirculation flow may result in approaching unstable reactor conditions and are not allowed to be used to comply with this Required Action. The 2 hour Completion Time is based on engineering judgement as to a reasonable time to restore the FCBB to within limit. The 2 hour Completion Time is acceptable based on the availability of the PBDS per Specification 3.3.1.3, "Period Based Detection System" and the low probability of a neutronic/thermal hydraulic instability event.

B.1 and B.2

Changes in reactor core state conditions resulting from an unexpected loss of feedwater heating or unexpected reduction in core flow (e.g., any unexpected reduction in feedwater temperature, recirculation pump trip, or recirculation pump run back) require immediate initiation of action to exit the Restricted Region and return the APRM Flow Biased Simulated Thermal Power—High Function (refer to LCO 3.3.1.1, Table 3.3.1.1-1, Function 2.b.) to the "non-Setup" value. Condition B is modified by a Note that specifies that Required Actions B.1 and B.2 must be completed if this Condition is entered due to an unexpected loss of feedwater heating or unexpected reduction in core flow. The completion of Required Actions B.1 and B.2 is required even though FCBB may be calculated and determined to be within limit. Core conditions continue to change after an unexpected loss of feedwater heating or unexpected reduction in core flow due to transient induced changes with the

(continued)

ACTIONS

B.1 and B.2 (continued)

potential that the FCBB may change and the limit not be met. The potential for changing core conditions, with FCBB not met, is not consistent with operation in the Restricted Region or with the APRM Flow Biased Simulated Thermal Power—High Function "Setup". Therefore, actions to exit the Restricted Region and return the APRM Flow Biased Simulated Thermal Power—High Function to the "non-Setup" value are required to be completed in the event Condition B is entered due to an unexpected loss of feedwater heating or an unexpected reduction in core flow.

If Operator actions to restore the FCBB to within limit are not successful within the specified Completion Time of Condition A, reactor operating conditions may be changing and may continue to change such that core conditions necessary for the onset of neutronic/thermal hydraulic instability may be met. Therefore, in the event the Required Action and associated Completion Time of Condition A is not met, immediate action to exit the Restricted Region and return the APRM Flow Biased Simulated Thermal Power—High Function to the "non-Setup" value is required.

Exit of the Restricted Region can be accomplished by control rod insertion and/or recirculation flow increases. Actions to restart an idle recirculation loop, withdraw control rods or reduce recirculation flow may result in approaching unstable reactor conditions and are not allowed to be used to comply with this Required Action. The time required to exit the Restricted Region will depend on existing plant conditions. Provided efforts are begun without delay and continued until the Restricted Region is exited, operation is acceptable.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

Verifying FCBB ≤ 1.0 is required to ensure the reactor is operating within the assumptions of the safety analysis. The boiling boundary limit is established to ensure that the core will remain stable during normal reactor operations in the Restricted Region of the power and flow map defined in the COLR which may otherwise be susceptible to neutronic/thermal hydraulic instabilities.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1 (continued)

FCBB is required to be verified every 24 hours while operating in the Restricted Region defined in the COLR. The 24 hour Frequency is based on both engineering judgment and recognition of the slow rate of change in power distribution during normal operation.

The second Frequency requires FCBB to be within the limit within 15 minutes following an unexpected transient. The verification of the FCBB is required as a result of the possibility that the unexpected transient results in the limit not being met. The 15 minute frequency is based on both engineering judgement and the availability of the PBDS to provide the operator with information regarding the potential imminent onset of neutronic/thermal hydraulic instability. The 15 minute Frequency for this SR is not to be used to delay entry into Condition B following an unexpected reduction in feedwater heating, recirculation pump trip, or recirculation pump run back.

This Surveillance is modified by a Note which allows 15 minutes to verify FCBB following entry into the Restricted Region if the entry was the result of an unexpected transient (i.e., an unintentional or unplanned change in core thermal power or core flow). The 15 minute allowance is based on both engineering judgement and the availability of the PBDS to provide the operator with information regarding the potential imminent onset of neutronic/thermal hydraulic instability. The 15 minute allowance of the Note is not to be used to delay entry into Condition B if the entry into the Restricted Region was the result of an unexpected reduction in feedwater heating, recirculation pump trip, or recirculation pump run back.

REFERENCES

1. NEDO 32339-A, Reactor Stability Long Term Solution: Enhanced Option I-A, July 1995.

2. 10 CFR 50.36(c)(2)(ii).

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux—High, Startup (continued)

this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 11 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux—High, Startup Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

In MODE 1, the Average Power Range Monitor Flow Biased Simulated Thermal Power—High and Fixed Neutron Flux—High Functions provide protection against reactivity transients and the RWM and Rod Block Monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power—High

The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of rated recirculation drive flow (W) in percent and is clamped at an upper limit that is always lower than the Average Power Range Monitor Fixed Neutron Flux—High Function Allowable Value. The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function provides a general definition of the licensed core power/core flow operating domain.

The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function is not associated with an LSSS. Operating limits established for the licensed operating domain are used to develop the Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function Allowable Values to provide pre-emptive reactor scram and prevent

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power—High (continued)

gross violation of the licensed operating domain. Operation outside the licensed operating domain may result in anticipated operational occurrences and postulated accidents being initiated from conditions beyond those assumed in the safety analysis. Operation within the licensed operating domain also ensures compliance with General Design Criterion 12.

General Design Criterion 12 requires protection of fuel thermal safety limits from conditions caused by neutronic/thermal hydraulic instability. Neutronic/thermal hydraulic instabilities result in power oscillations which could result in exceeding the MCPR SL.

The area of the core power and flow operating domain susceptible to neutronic/thermal hydraulic instability is affected by the Fraction of Core Boiling Boundary (LCO 3.2.3, FCBB). "Setup" and normal ("non-Setup") Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function Allowable Values are specified in the Core Operating Limits Report (COLR). When the normal value is selected protection against neutronic/thermal hydraulic instability is provided by preventing operation in the susceptible area of the operating domain during operation outside the Restricted Region of the operating domain specified in the COLR with the FCBB limit not required to be met. When the "Setup" value is selected meeting the FCBB limit provides protection against neutronic/thermal hydraulic instability.

"Setup" and "non-Setup" values are selected by operator manipulation of the recessed Setup button on each flow control trip reference card. Selection of the "Setup" value is intended only for planned operation in the Restricted Region as specified in the COLR. Operation in the Restricted Region with the Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function in "Setup" requires the FCBB limit to be met and is not generally consistent with normal power operation.

The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function uses a trip level generated by the flow control trip reference card based on recirculation loop drive flow. The proper representation of drive flow as

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power—High (continued)

indication of core flow is ensured through drive flow alignment. This is accomplished by the selection of appropriate dip switch positions on the flow control trip reference cards (Refer to SR 3.3.1.1.18). Changes in the core flow to drive flow functional relationship may vary over the core flow operating range. These changes can result from gradual changes in the Recirculation System and core components over the reactor life time as well as specific maintenance performed on these components (e.g., jet pump cleaning).

The APRM System is divided into two groups of channels with three APRM inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Flow Biased Simulated Thermal Power—High with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 11 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located. Each APRM channel receives two total drive flow signals representative of total core flow. The total drive flow signals are generated by four flow units, two of which supply signals to the trip system A APRMs, while the other two supply signals to the trip system B APRMs. Each flow unit signal is provided by summing up the flow signals from the two recirculation loops. To obtain the most conservative reference signals, the total flow signals from the two flow units (associated with a trip system as described above) are routed to a low auction circuit associated with each APRM. Each APRM's auction circuit selects the lower of the two flow unit signals for use as the scram trip reference for that particular APRM. Each required Average Power Range Monitor Flow Biased Simulated Thermal Power—High channel only requires an input from one OPERABLE flow unit, since the individual APRM channel will perform the intended function with only one OPERABLE flow unit input. However, in order to maintain single failure criteria as described above for the Function, at least one required Average Power Range

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power—High (continued)

Monitor Flow Biased Simulated Thermal Power—High channel in each trip system must be capable of maintaining an OPERABLE flow unit signal in the event of a failure of an auction circuit, or a flow unit, in the associated trip system (e.g., if a flow unit is inoperable, one of the two required Average Power Range Monitor Flow Biased Simulated Thermal Power—High channels in the associated trip system must be considered inoperable).

The THERMAL POWER time constant of ≤ 7 seconds is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER.

The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function is required to be OPERABLE in MODE 1 when there is the possibility of neutronic/thermal hydraulic instability. The potential to exceed the SL applicable to high pressure and core flow conditions (MCPR SL), which provides fuel cladding integrity protection, exists if neutronic/thermal hydraulic instability occurs. During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

2.c. Average Power Range Monitor Fixed Neutron Flux—High

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The Average Power Range Monitor Fixed Neutron Flux—High Function is capable of generating a trip signal without the electronically filtered time constant to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of References 4 and 7, the Average Power Range Monitor Fixed Neutron Flux—High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (SRVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 2) takes credit for the Average Power Range Monitor Fixed Neutron Flux—High Function to terminate the CRDA.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.11 and SR 3.3.1.1.13 (continued)

calibration (SR 3.3.1.1.3) and the 1100 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.8). A second Note is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR. Note 3 to SR 3.3.1.1.11 states that the digital components of the flow control trip reference card are excluded from CHANNEL CALIBRATION of Function 2.b, Average Power Range Monitor Flow Biased Simulated Thermal Power--High. The analog output potentiometers of the flow control trip reference card are not excluded from this test. The flow control trip reference card has an automatic self-test feature which periodically tests the hardware that performs the digital algorithm. Exclusion of the digital components of the flow control trip reference card from CHANNEL CALIBRATION of Function 2.b is based on conditions required to perform the test and the small likelihood of a change in the status of these components not being detected.

The Frequency of SR 3.3.1.1.11 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.13 is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.1.14

The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function uses an electronic filter circuit to generate a signal proportional to the core THERMAL POWER from the APRM neutron flux signal. This filter circuit is representative of the fuel heat transfer dynamics that produce the relationship between the neutron flux and the core THERMAL POWER. The filter time constant must be verified to be $\leq$ 7 seconds to ensure that the channel is accurately reflecting the desired parameter.

The Frequency of 24 months is based on engineering judgment considering the reliability of the components.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.17

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. This test may be performed in one measurement or in overlapping segments, with verification that all components are tested. The RPS RESPONSE TIME acceptance criteria are included in Reference 13.

As noted (Note 1), neutron detectors for Function 2 are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time. In addition, Note 2 states the response time of the sensors for Functions 3 and 4 may be assumed in the RPS RESPONSE TIME test to be the design sensor response time. This is allowed since the sensor response time is a small part of the overall RPS RESPONSE TIME (Ref. 14).

RPS RESPONSE TIME tests are conducted on a 24 month STAGGERED TEST BASIS. Note 3 requires STAGGERED TEST BASIS Frequency to be determined based on 4 channels per trip system, in lieu of the 8 channels specified in Table 3.3.1.1-1 for the MSIV Closure Function. This Frequency is based on the logic interrelationships of the various channels required to produce an RPS scram signal. The 24 month Frequency is consistent with the BNP refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

SR 3.3.1.1.18

The Average Power Range Monitor Flow Biased Simulated Thermal Power—High Function uses a trip level generated by the flow control trip reference card based on the recirculation loop drive flow. The drive flow is adjusted by a digital algorithm according to selected drive flow alignment dip switch settings. This SR sets, as necessary, the flow control trip reference card to ensure the drive flow alignment used results in the appropriate trip level being generated from the digital components of the card.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.18 (continued)

The Frequency of once within 7 days after reaching equilibrium conditions following a refueling outage is based on the expectation that any change in the core flow to drive flow functional relationship during power operation would be gradual and the maintenance on the Recirculation System and core components which may impact the relationship is expected to be performed during refueling outages. The 7 day time period to reach equilibrium conditions is based on plant conditions required to perform the test, engineering judgment of the time required to collect and analyze the necessary flow data, and engineering judgment of the time required to adjust and check the adjustment of each flow control trip reference card. The 7 day time period to reach equilibrium conditions is acceptable based on the low probability of a neutronic/thermal hydraulic instability event.

REFERENCES

1. UFSAR, Section 7.2.
2. UFSAR, Chapter 15.0.
3. UFSAR, Section 7.2.2.
4. NEDC-32466P, Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2, September 1995.
5. 10 CFR 50.36(c)(2)(ii).
6. NEDO-23842, Continuous Control Rod Withdrawal in the Startup Range, April 18, 1978.
7. UFSAR, Section 5.2.2.
8. UFSAR, Appendix 5.2A.
9. UFSAR, Section 6.3.1.
10. P. Check (NRC) letter to G. Lainas (NRC), BWR Scram Discharge System Safety Evaluation, December 1, 1980.
11. NEDC-30851-P-A, Technical Specification Improvement Analyses for BWR Reactor Protection System, March 1988.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.1.3 Period Based Detection System (PBDS)

BASES

BACKGROUND

General Design Criteria 12 requires protection of fuel thermal safety limits from conditions caused by neutronic/thermal hydraulic instability. Neutronic/thermal hydraulic instabilities can result in power oscillations which could result in exceeding the MCPR Safety Limit (SL). The MCPR SL ensures that at least 99.9% of the fuel rods avoid boiling transition during normal operation and during an anticipated operational occurrence (AOO) (refer to the Bases for SL 2.1.1.2).

The PBDS provides the operator with an indication that conditions consistent with a significant degradation in the stability performance of the reactor core has occurred and the potential for imminent onset of neutronic/thermal hydraulic instability may exist. Indication of such degradation is cause for the operator to initiate an immediate reactor scram if the reactor is being operated in either the Restricted Region or Monitored Region. The Restricted Region and Monitored Region are defined in the COLR.

The PBDS instrumentation of the Neutron Monitoring System (NMS) consists of two channels. PBDS channel A includes input from 13 local power range monitors (LPRMs) within the reactor core and PBDS channel B includes input from 11 LPRMs within the reactor core. All LPRMs are utilized from each of the axial levels except for the D level detectors. These inputs are continually monitored by the PBDS for variations in the neutron flux consistent with the onset of neutronic/thermal hydraulic instability. Each channel includes separate local indication and separate control room High—High Alarms. While, this LCO specifies OPERABILITY requirements only for one monitoring and indication channel of the PBDS, if both are OPERABLE, a High—High Alarm from either channel results in the need for the operator to take actions.

The primary PBDS component is a card in the NMS with analog inputs and digital processing. The PBDS card has an automatic self-test feature to periodically test the hardware circuit. The self-test functions are executed during their allocated portion of the executive loop

(continued)

BASES

BACKGROUND (continued)

sequence. Any self-test failure indicating loss of critical function results in a common control room "Inoperative" alarm. The inoperable condition is also displayed by an indicating light on the card front panel. A manually initiated internal test sequence can be actuated via a recessed push button. This internal test consists of simulating alarm and inoperable conditions to verify card OPERABILITY. Further descriptions of the PBDS are provided in References 1 and 2.

Actuation of the PBDS High—High Alarm is not postulated to occur due to neutronic/thermal hydraulic instability during operation outside the Restricted Region and the Monitored Region. Periodic perturbations can be introduced into the thermal hydraulic behavior of the reactor core from external sources such as recirculation system components and the pressure and feedwater control systems. These perturbations can potentially drive the neutron flux to oscillate within a frequency range expected for neutronic/thermal hydraulic instability. The presence of such oscillations may be recognized by the period based algorithm of the PBDS and could result in a High—High Alarm. Actuation of the PBDS High—High Alarm outside the Restricted Region and the Monitored Region indicate the presence of a source external to the reactor core and are not indications of neutronic/thermal hydraulic instability.

APPLICABLE SAFETY ANALYSES

Analysis, as described in Section 4 of Reference 1, confirms that AOOs initiated from outside the Restricted Region without stability control and from within the Restricted Region with stability control are not expected to result in neutronic/thermal hydraulic instability. The stability control applied in the Restricted Region (refer to LCO 3.2.3, "Fraction of Core Boiling Boundary (FCBB)") is established to prevent neutronic/thermal hydraulic instability during operation in the Restricted Region. Operation in the Monitored Region is only susceptible to instability under operating conditions beyond those analyzed in Reference 1. The types of transients specifically evaluated are loss of flow and coolant temperature decrease which are limiting for the onset of instability.

The initial conditions assumed in the analysis are reasonably conservative and the immediate post-event reactor conditions are significantly stable. However, these assumed initial conditions do not bound each individual parameter

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

which impacts stability performance (Ref. 1). The PBDS instrumentation provides the operator with an indication that conditions consistent with a significant degradation in the stability performance of the reactor core has occurred and the potential for imminent onset of neutronic/thermal hydraulic instability may exist. Such conditions are only postulated to result from events initiated from initial conditions beyond the conditions assumed in the safety analysis (refer to Section 4, Ref. 1).

The PBDS has no safety function and is not assumed to function during any UFSAR design basis accident or transient analysis. However, the PBDS provides the only indication of the imminent onset of neutronic/thermal hydraulic instability during operation in regions of the operating domain potentially susceptible to instability. Therefore, the PBDS is included in the Technical Specifications.

LCO

One PBDS channel is required to be OPERABLE with a minimum of eight LPRM inputs to monitor reactor neutron flux for indications of imminent onset of neutronic/thermal hydraulic instability. A PBDS channel may be considered OPERABLE with six LPRM inputs when the distribution of OPERABLE LPRMs provides: a) at least one OPERABLE LPRM in each core quadrant or b) at least two OPERABLE LPRMs in the core quadrant opposite any core quadrant with no OPERABLE LPRMs. The required distribution of the LPRMs when a PBDS channel is considered OPERABLE with as few as six OPERABLE LPRMs ensures a minimum of two OPERABLE LPRMs in opposite core quadrants. This distribution ensures that, for all postulated orientations and modes of oscillation, there are at least two OPERABLE LPRMs in the core quadrants in which the local neutron flux will oscillate with a frequency within the range monitored by the PBDS. OPERABILITY requires the ability for the operator to be immediately alerted to a High—High Alarm. This is accomplished by the instrument channel control room alarm. The LCO also requires reactor operation be such that the High—High Alarm is not actuated by any OPERABLE PBDS instrumentation channel.

APPLICABILITY

At least one of two PBDS instrumentation channels is required to be OPERABLE during operation in either the Restricted Region or the Monitored Region specified in the

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BASES

APPLICABILITY (continued)

COLR. Similarly, operation with the PBDS High—High Alarm of any OPERABLE PBDS instrumentation channel is not allowed in the Restricted Region or the Monitored Region. Operation in these regions is susceptible to instability (refer to the Bases for LCO 3.2.3 and Section 4 of Ref. 1). OPERABILITY of at least one PBDS instrumentation channel and operation with no indication of a PBDS High—High Alarm from any OPERABLE PBDS instrumentation channel is therefore required during operation in these regions.

The boundary of the Restricted Region in the Applicability of this LCO is analytically established in terms of thermal power and core flow. The Restricted Region is defined by the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints, which are a function of reactor recirculation drive flow. The Restricted Region Entry Alarm (RREA) signal is generated by the Flow Control Trip Reference (FCTR) card using the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints. As a result, the RREA is coincident with the Restricted Region boundary under all anticipated operating conditions when the setpoints are not "Setup," and provides the indication regarding entry into the Restricted Region. However, APRM Flow Biased Simulated Thermal Power—High Control Rod Block signals provided by the FCTR card, that are not coincident with the Restricted Region boundary, do not generate a valid RREA. The Restricted Region boundary for this LCO Applicability is specified in the COLR.

When the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints are "Setup" the applicable setpoints used to generate the RREA are moved to the interior boundary of the Restricted Region to allow controlled operation within the Restricted Region. While the setpoints are "Setup" the Restricted Region boundary remains defined by the normal APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints. Parameters such as reactor power and core flow available at the reactor controls, may be used to provide immediate confirmation that entry into the Restricted Region could reasonably have occurred. The Monitored Region in the Applicability of this LCO is analytically established in terms of thermal power and core flow. However, unlike the Restricted Region boundary the Monitored Region is not specifically monitored by plant instrumentation to provide automatic indication of entry into the region. Therefore, the Monitored Region

(continued)

BASES

APPLICABILITY (continued)

boundary is defined solely in terms of thermal power and core flow. The Monitored Region boundary for this LCO Applicability is specified in the COLR.

Operation outside the Restricted Region and the Monitored Region is not susceptible to neutronic/thermal hydraulic instability even under extreme postulated conditions.

ACTIONS

A.1

If at any time while in the Restricted Region or Monitored Region, an OPERABLE PBDS instrumentation channel indicates a High—High Alarm, the operator is required to initiate an immediate reactor scram. Verification that the High—High Alarm is valid may be performed without delay against another output from a PBDS card observable from the reactor controls in the control room prior to the manual reactor scram. This provides assurance that core conditions leading to neutronic/thermal hydraulic instability will be mitigated. This Required Action and associated Completion Time does not allow for evaluation of circumstances leading to the High—High Alarm prior to manual initiation of reactor scram.

B.1 and B.2

Operation with the APRM Flow Biased Simulated Thermal Power—High Function (refer to LCO 3.3.1.1, Table 3.3.1.1-1, Function 2.b) "Setup" requires the stability control applied in the Restricted Region (refer to LCO 3.2.3) to be met. Requirements for operation with the stability control met are established to prevent reactor thermal hydraulic instability during operation in the Restricted Region. When the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints are not "Setup" uncontrolled entry into the Restricted Region is identified by receipt of a valid RREA. Immediate confirmation that the RREA is valid and indicates an actual entry into the Restricted Region may be performed without delay. Immediate confirmation constitutes observation that plant parameters immediately available at the reactor controls (e.g., core power and core flow) are reasonably consistent with entry into the Restricted Region. This immediate confirmation may also constitute recognition that plant parameters are rapidly changing during a

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

transient (e.g., a recirculation pump trip) which could reasonably result in entry into the Restricted Region. While the APRM Flow Biased Simulated Thermal Power—High Control Rod Block setpoints are "Setup," operation in the Restricted Region may be confirmed by use of plant parameters such as reactor power and core flow available at the reactor controls. With the required PBDS channel inoperable while in the Restricted Region, the ability to monitor conditions indicating the potential for imminent onset of neutronic/thermal hydraulic instability as a result of unexpected transients is lost. Therefore, action must be immediately initiated to exit the Restricted Region.

Exit of the Restricted Region can be accomplished by control rod insertion and/or recirculation flow increases. Actions to restart an idle recirculation loop, withdraw control rods or reduce recirculation flow may result in unstable reactor conditions and are not allowed to be used to comply with this Required Action.

The time required to exit the Restricted Region will depend on existing plant conditions. Provided efforts are begun without delay and continued until the Restricted Region is exited, operation is acceptable based on the low probability of a transient which degrades stability performance occurring simultaneously with the required PBDS channel inoperable.

Required Action B.1 is modified by a Note that specifies that initiation of action to exit the Restricted Region only applies if the APRM Flow Biased Simulated Thermal Power—High Function is "Setup". Operation in the Restricted Region without the APRM Flow Biased Simulated Thermal Power—High Function "Setup" indicates uncontrolled entry into the Restricted Region. Uncontrolled entry is consistent with the occurrence of unexpected transients, which, in combination with the absence of stability controls being met may result in significant degradation of stability performance. Under these conditions with the required PBDS instrumentation channel inoperable, the ability to monitor conditions indicating the potential for imminent onset of neutronic/thermal hydraulic instability is lost and continued operation is not justified. Therefore, Required Action B.2 requires immediate reactor scram.

(continued)

BASES

ACTIONS (continued)

C.1

In the Monitored Region the PBDS High—High Alarm provides indication of degraded stability performance. Although not anticipated, operation in the Monitored Region is susceptible to neutronic/thermal hydraulic instability under postulated conditions exceeding those previously assumed in the safety analysis. With the required PBDS channel inoperable while in the Monitored Region, the ability to monitor conditions indicating the potential for imminent onset of neutronic/thermal hydraulic instability is lost. Therefore, action must be initiated to exit the Monitored Region.

Actions to restart an idle recirculation loop, withdraw control rods or reduce recirculation flow may result in approaching unstable reactor conditions and are not allowed to be used to comply with this Required Action. Exit of the Monitored Region is accomplished by control rod insertion and/or recirculation flow increases. However, actions which reduce recirculation flow are allowed provided the FCBB is recently (within 15 minutes) verified to be ≤ 1.0. Recent verification of FCBB being met, provides assurance that with the PBDS inoperable, planned decreases in recirculation drive flow should not result in significant degradation of core stability performance.

The Completion Time of 15 minutes ensures timely operator action to exit the region consistent with the low probability that reactor conditions exceed the initial conditions assumed in the safety analysis. The time required to exit the Monitored Region will depend on existing plant conditions. Provided efforts are begun within 15 minutes and continued until the Monitored Region is exited, operation is acceptable based on the low probability of a transient which degrades stability performance occurring simultaneously with the required PBDS channel inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3.1

During operation in the Restricted Region or the Monitored Region the PBDS High—High Alarm is relied upon to indicate conditions consistent with the onset of neutronic/thermal hydraulic instability. Verification that each OPERABLE

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SURVEILLANCE REQUIREMENTS

SR 3.3.1.3.1 (continued)

channel of PBDS instrumentation is not in High—High Alarm every 12 hours provides assurance of the proper indication of the alarm during operation in the Restricted Region or the Monitored Region. The 12 hour Frequency supplements less formal, but more frequent, checks of alarm status during operation.

SR 3.3.1.3.2

Performance of the CHANNEL CHECK every 12 hours ensures that a gross failure of instrumentation has not occurred. This CHANNEL CHECK is normally a comparison of the PBDS indication to the state of the annunciator, as well as comparison to the same parameter on the other channel if it is available. It is based on the assumption that the instrument channel indication agrees with the immediate indication available to the operator, and that instrument channels monitoring the same parameter should read similarly. Deviations between the instrument channels could be an indication of instrument component failure. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL FUNCTIONAL TEST. Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability.

The 12 hour Frequency is based on the CHANNEL CHECK Frequency requirement of similar Neutron Monitoring System components. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.3.3

A CHANNEL FUNCTIONAL TEST is performed for each required PBDS channel to ensure that the system will perform the intended function. The CHANNEL FUNCTIONAL TEST for the PBDS includes manual initiation of an internal test sequence and verification of appropriate alarms and inop conditions being reported.

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BASES

SURVEILLANCE REQUIREMENTS	SR 3.3.1.3.3 (continued) Performance of a CHANNEL FUNCTIONAL TEST at a Frequency of 24 months verifies the performance of the PBDS and associated circuitry. The Frequency considers the plant conditions required to perform the test, the ease of performing the test, and the likelihood of a change in the system or component status. The alarm circuit is designed to operate for over 24 months with sufficient accuracy on signal amplitude and signal timing considering environment, initial calibration, and accuracy drift (Ref. 2).
REFERENCES	1. NEDO 32339-A, Reactor Stability Long Term Solution: Enhanced Option I-A, July 1994. 2. NEDC-32339, Supplement 2, Reactor Stability Long Term Solution: Enhanced Option I-A Solution Design. April 1995.