SURVEILLANCE REQUIREMENTS

--------------------NOTES--------------------

[4.1.A] 1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

[3.1-1 nt 2] 2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

	SURVEILLANCE	FREQUENCY
[M5,M6] [T.4.1-1] SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
[T.4.1-2] [M10, L12, L13] SR 3.3.1.1.2	-----NOTE----- Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. ----- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP ~~[plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Setpoints"]~~ while operating at ≥ 25% RTP. Gain and 1	7 days CLB1 J CLB2
~~SR 3.3.1.1.3~~	~~Adjust the channel to conform to a calibrated flow signal.~~	~~7 days~~
[L10] [T.4.1-1] SR 3.3.1.1.~~4~~ 3 CLB2	-----NOTE----- Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ----- Perform CHANNEL FUNCTIONAL TEST.	7 days

(continued)

Revision J

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
[T. 4.1-1]	SR 3.3.1.1.~~5~~ 4 CLB2	Perform CHANNEL FUNCTIONAL TEST. CLB3 a functional test of each RPS automatic scram contactor	7 days	
[M7]	SR 3.3.1.1.~~6~~ 5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to fully withdrawing SRMs ~~from the fully inserted position~~	DB10 J
[T 4.1-2] [M14]	SR 3.3.1.1.~~7~~ 6 CLB2	-------------------NOTE------------------- Only required to be met during entry into MODE 2 from MODE 1. --- Verify the IRM and APRM channels overlap.	7 days	
[T. 4.1.2]	SR 3.3.1.1.~~8~~ 7 CLB2	Calibrate the local power range monitors.	1000 MWD/T average core exposure	
[T 4.1-1]	SR 3.3.1.1.~~9~~ 8 CLB2	Perform CHANNEL FUNCTIONAL TEST.	~~[92]~~ days CLB4	
[T 4.1-2, Note 6]	SR 3.3.1.1.10	Calibrate the trip units.	~~[92]~~ 184 days	CLB5

(continued)

Insert SR 3.3.1.1.9 from next page

DB7

Revision J

1J

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

PA2

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level — High		DB3		DB5	
[T. 3.3-1 (12)] [T.4.1-1(14)] [T.4.1-2(9)] a. ~~Resistance Temperature Detector~~ Differential Pressure Transmitter / Trip unit DB4	1,2	~~2~~	G	SR 3.3.1.1.1 SR 3.3.1.1.9 ~~SR 3.3.1.1.10~~ ~~SR 3.3.1.1.13~~ SR 3.3.1.1.13	≤ ~~57.15~~ 34.5 gallons DB9
CLB3	5(a)	~~2~~	H	SR 3.3.1.1.1 SR 3.3.1.1.9 DB5 ~~SR 3.3.1.1.10~~ ~~SR 3.3.1.1.13~~ SR 3.3.1.1.13	≤ ~~57.15~~ gallons
SR 3.3.1.1.4					
[T.4.1-2(8)] [T. 3.3-1 (12)] [T.4.1-1(13)] b. ~~Float~~ Level Switch PA1	1,2	~~2~~	G	SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.13	≤ ~~57.15~~ 34.5 gallons DB9
	5(a)	~~2~~	H	SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.13	≤ ~~57.15~~ gallons
[T.4.1-1 (17) & (18)] [T.4.1-2(11) & (13)] [T. 3.3-1(15)] [2.1.A.3] [4.1.A.5] [L14] 8. Turbine Stop Valve — Closure DB1	≥ ~~30~~ 29% RTP	~~4~~	E	SR 3.3.1.1.8 ~~SR 3.3.1.1.10~~ 8 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14 19 SR 3.3.1.1.15	≤ ~~10~~% closed 15
[T.4.1-1(16) & (17)] [T.4.1-2(11) & 12] [T. 3.3-1(14)] [2.1.A.4] [4.1.A.6] [L14] 9. Turbine Control Valve Fast Closure, ~~Trip~~ EHC Oil Pressure — Low PA1	≥ ~~30~~ 29% RTP	~~2~~ DB6 SR 3.3.1.1.1	E	SR 3.3.1.1.8 ~~SR 3.3.1.1.10~~ 8 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ ~~600~~ 500 psig and ≤ 850 psig DB9
[T. 3.3-1 (1)] [T. 4.1-1(1)] 10. Reactor Mode Switch — Shutdown Position	1,2	~~2~~ 1	G	SR 3.3.1.1.11 SR 3.3.1.1.13	NA
	5(a)	~~2~~ 1	H	SR 3.3.1.1.11 SR 3.3.1.1.13	NA
[T. 3.3-1 (2)] [T.4.1-1 (2)] 11. Manual Scram	1,2	~~2~~ 1	G	SR 3.3.1.1.8 SR 3.3.1.1.13	NA
	5(a)	~~2~~ 1 DB3	H	SR 3.3.1.1.8 SR 3.3.1.1.13	NA

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

Revision J

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The brackets in SR 3.3.1.1.2 have been removed and the plant specific requirements included in accordance with CTS 4.1.B.

CLB2 ISTS SR 3.3.1.1.3, the requirement to adjust the channels to conform to a calibrated signal every 7 days has been deleted since this requirement is currently being performed along with the 92 day channel functional test. This adjustment will be performed in accordance with SR 3.3.1.1.8, the 92 day CHANNEL FUNCTIONAL TEST. This is reflected in the Bases of SR 3.3.1.1.8. Subsequent SRs have been renumbered, as applicable.

CTS 4.1.2 "Flow Biased Signal" requires an "internal power and flow test with standard pressure source" calibration on a "refueling interval," which has been translated into ITS SR 3.3.1.1.12. This calibration of the flow signal is at a frequency that is consistent with the current licensing basis. The Functional Test of the APRMs (ITS SR 3.3.1.1.8) is consistent with CTS Table 4.1-1, which ensures the APRM circuitry responds appropriately to this calibrated flow signal. As such, the proposed ITS adequately translates the current licensing basis for testing the APRM Flow Biased Function without adopting the ISTS SR 3.3.1.1.3. In addition, since ITS SR 3.3.1.1.9, the 92 day SR, also applies to the Neutron Flux—High (Flow Biased) Function, Notes have been added to ensure SR 3.3.1.1.12 only applies to the recirculation loop flow signal portion of the channel and SR 3.3.1.1.9 applies to the remaining portions of the channel.

CLB3 SR 3.3.1.1.4 has been revised in accordance with CTS Table 4.1-1 and Note 1. This functional test was added to allow surveillance test interval extensions of the automatic RPS Functions per NEDC-30851-P-A, Technical Specification Improvement Analyses for BWR Reactor Protection System, since the JAFNPP design is different than the generic BWR model used in NEDC-30851-P-A. Therefore, it is associated with each automatic RPS Function in Table 3.3.1.1-1.

CLB4 The brackets have been removed for the Frequency of ISTS SR 3.3.1.1.9 (ITS SR 3.3.1.1.8) and the 92 day Frequency retained consistent with CTS Table 4.1-1 and with the reliability analysis of NEDC-30851-P-A.

CLB5 SR 3.3.1.1.10 Surveillance Frequency has been modified to be consistent with the frequency in CTS Table 4.1-2 Note 6 and approved in JAFNPP Technical Specification Amendment No. 89.

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB6 The brackets have been removed from the CHANNEL FUNCTIONAL TEST Frequency in ITS SR 3.3.1.1.11 and extended from 18 months to 24 months consistent with the Channel Functional Test frequencies of CTS Table 4.1-1. The Frequency is consistent with the JAFNPP fuel cycle.

CLB7 Not Used.

CLB8 Table 3.3.1.1-1 Function 2.d has been deleted, since the Downscale trip has been removed from the CTS as documented in JAFNPP License Amendment 227. The following Function has been renumbered as required.

CLB9 Table 3.3.1.1-1 Function 6, SR 3.3.1.1.16 RPS Response Time Surveillance requirements have been added consistent with CTS 4.1.A.2.

CLB10 Note 3 of ITS SR 3.3.1.1.15 has been changed to ensure that all channels are tested within two surveillance intervals consistent with the current licensing basis. In addition, the bracketed SR Frequency has been changed from 18 to 24 months consistent with the current Frequency in CTS 4.1.A.

CLB11 Not used.

CLB12 The Allowable Value for Function 2.b, APRM Neutron Flux—High (Flow Biased) is specified in the COLR.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 The Specification has been modified to reflect plant specific nomenclature.

PA2 The SRs associated with each Function in Table 3.3.1.1-1 have been renumbered as required, consistent with changes to the ITS 3.3.1.1 SURVEILLANCE REQUIREMENTS Table. Any specific change not reflected in the SURVEILLANCE REQUIREMENTS Table is identified with a specific JFD.

PA3 Editorial correction made to be consistent with the format requirements of the ISTS.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the proper plant specific THERMAL POWER level has been included consistent with the analysis assumptions.

DB2 ISTS SR 3.3.1.1.14 has been deleted because the JAFNPP RPS design does not include the APRM Flow Biased Simulated Thermal Power-High Function (time constant). Subsequent SRs have been renumbered, where applicable. In addition, Function 2.b has been renamed accordingly.

DB3 The brackets have been removed and the proper number of channels included for each Function in Table 3.3.1.1-1. The values are consistent with the current requirements in CTS Table 3.1-1 except for Functions 7.a, 7.b and 5. The number of channels for Functions 7.a, 7.b and 5 have been changed consistent with the plant design and justified in M2 and M3.

DB4 The plant specific device has been included for Function 7.a consistent with the current design.

DB5 For Function 7.a, ITS SR 3.3.1.1.10, the calibration of the trip unit, and ITS SR 3.3.1.1.12, the CHANNEL CALIBRATION test every 18 months, has been deleted since this Function is calibrated in accordance with ITS SR 3.3.1.1.9 every 92 days. Since this calibration includes the entire channel this specific requirement to calibrate the trip units, is not necessary. The 92 day CHANNEL CALIBRATION Frequency is consistent with the methodology for the setpoint calculation of this Function.

DB6 SR 3.3.1.1.1 has been included in Table 3.3.1.1-1 for Functions 8 and 9, to verify the turbine first stage pressure signal consistent with CTS Table 4.1-1.

DB7 ITS SR 3.3.1.1.9 has been added to perform a CHANNEL CALIBRATION every 92 days for Function 7.a (Scram Discharge Volume Water Level – High, Differential Pressure Transmitter/Trip Unit) consistent with CTS Table 4.1-2. The Frequency is consistent with the setpoint calculation methodology for this Function. In addition, the Frequency for ISTS SR 3.3.1.1.11, the 184 day CHANNEL CALIBRATION requirement for the APRM Functions, has been changed to 92 days (ITS SR 3.3.1.1.9), consistent with the CTS. Also, the IRMs are currently required to be tested every 92 days. Therefore, the Note to ISTS SR 3.3.1.1.13 has been incorporated into ITS SR 3.3.1.1.9, the 92 day CHANNEL CALIBRATION requirement. J

DB8 The brackets have been removed from the Surveillance Frequency in ITS SR 3.3.1.1.12 (CHANNEL CALIBRATION) and extended from 18 months to 24 months consistent with the frequencies in CTS Table 4.1-2 and as

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB8 (continued)

justified in M9 for the IRM High Flux channels. The Frequency is consistent with the setpoint calculation methodology for the associated Functions.

DB9 The brackets have been removed and the proper plant specific "Allowable Value" has been included consistent with the current value in CTS Table 3.1-1, and the JAFNPP plant specific setpoints methodology. Footnote b of ITS Table 3.3.1.1-1 has been deleted since the Flow Biased Setpoint is included in the COLR.

DB10 The Frequency for ISTS SR 3.3.1.1.6 (ITS SR 3.3.1.1.5) has been changed from "Prior to withdrawing SRMs from the fully inserted position" to "Prior to fully withdrawing SRMs." The current licensing basis for JAFNPP does not require the SRM/IRM overlap to be verified. The current practice of JAFNPP is to maintain the SRMs between 100 cps and 10^5 cps. Thus, during the reactor startup, the operating staff will normally start to withdraw the SRMs prior to the ITS SRMs/IRMs overlap requirement being met. This reduces the burnup of the SRMs. The SRM/IRM overlap is verified before the SRMs are fully withdrawn. In addition, a review of operating data has shown that it may not always be possible to obtain proper overlap prior to reaching the SRM rod block setpoint with the SRMs fully inserted. Therefore, ITS SR 3.3.1.1.5 has been modified to reflect the current practice.

J

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 332, Revision 1 have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 The brackets have been removed from the Frequency in ITS SR 3.3.1.1.13 (the LOGIC SYSTEM FUNCTIONAL TEST) and the 18 month surveillance extended to 24 months as justified in M4. This Frequency is consistent with the JAFNPP fuel cycle.

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X2 The brackets have been removed from the Frequency in ITS SR 3.3.1.1.14 (the verification bypass feature) and the 18 month surveillance extended to 24 months as justified in M13. This Frequency is consistent with the JAFNPP fuel cycle.

B 3.3 INSTRUMENTATION

B 3.3.1.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor scram when one or more monitored parameters exceed their specified limits, to preserve the integrity of the fuel cladding and the Reactor ~~Coolant System (RCS)~~ pressure boundary (RCPB) and minimize the energy that must be absorbed following a loss of coolant accident (LOCA). This can be accomplished either automatically or manually. PA2

The protection and monitoring functions of the RPS have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance. ~~The LSSS are defined in this Specification as the Allowable Values, which, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits, including Safety Limits (SLs) during Design Basis Accidents (DBAs).~~ PA3 TA4 Insert BKGD-2 DB2

PA2 PA2 PA1

The RPS, as ~~shown~~ described in the uFSAR, ~~Figure 7.2-1~~ Section 7.2 (Ref. 1), includes sensors, relays, logic circuits, bypass circuits, and switches that are necessary to cause initiation of a reactor scram. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. The input parameters to the scram logic are from instrumentation that monitors reactor vessel water level, reactor vessel pressure, neutron flux, main steam line isolation valve position, turbine control valve (TCV) fast closure, ~~trip oil pressure,~~ EHC Oil Pressure-Low, turbine stop valve (TSV) position, drywell pressure, and scram discharge volume (SDV) water level, as well as reactor mode switch in shutdown position and manual scram signals. There are at least four redundant sensor input signals from each of these parameters (with the exception of the reactor mode switch in shutdown position and manual scram signals). Most channels include ~~electronic equipment (e.g., trip units)~~ instrumentation that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel ~~output relay actuates, which then~~ outputs an RPS trip signal to the trip logic. ~~Table B 3.3.1.1-1 summarizes the diversity of sensors capable of initiating scrams during anticipated operating transients typically analyzed.~~ PA1 PA3 J J DB1 DB1 DB1 DB1 PA4

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

the containment by minimizing the energy that must be absorbed following a LOCA.

RPS instrumentation satisfies Criterion 3 of ~~the NRC Policy Statement~~. Functions not specifically credited in the accident analysis are retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

X1

10 CFR 50.36 (c)(2)(ii) (Ref. 4)

The OPERABILITY of the RPS is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.1.1-1. Each Function must have a required number of OPERABLE channels per RPS trip system, with their setpoints within the specified Allowable Value, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Each channel must also respond within its assumed response time.

DB4

, where appropriate

Allowable Values are specified for ~~each~~ RPS Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

, as appropriate,

DB4

5

1 5

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. ~~The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe~~

DB1

or other appropriate documents

INSERT ASA

DB5

(continued)

Revision 5

DB3 INSERT Function 2.b-1

The flow biased Allowable Value is credited in the safety analyses (thermal-hydraulic instability) and is specifically confirmed for each operating cycle. For this reason the Allowable Value is included in the COLR for both single and two recirculation loop operation. The clamped portion of the Allowable Value is set more conservative than the APRM Neutron Flux–High (Fixed) (Function 2.c). 5

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

PA1

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor (Fixed) Neutron Flux—High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor (Fixed) Neutron Flux—High Function is assumed in the CRDA analysis (Ref. 8) which is applicable in MODE 2, the Average Power Range Monitor Neutron Flux—High (Startup) ~~Setdown~~ Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Range Monitor (Fixed) Neutron Flux—High Function is not required in MODE 2.

PA1 PA2 PA1 J PA1

CLB1

~~2.d. Average Power Range Monitor—Downscale~~

~~This signal ensures that there is adequate Neutron Monitoring System protection if the reactor mode switch is placed in the run position prior to the APRMs coming on scale. With the reactor mode switch in run, an APRM downscale signal coincident with an associated Intermediate Range Monitor Neutron Flux—High or Inop signal generates a trip signal. This Function was not specifically credited in the accident analysis but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.~~

~~The APRM System is divided into two groups of channels with three inputs into each trip system. The system is designed to allow one channel in each trip system to be bypassed. Four channels of Average Power Range Monitor—Downscale 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 failure will preclude a scram from this Function on a valid signal. The Intermediate Range Monitor Neutron Flux—High and Inop Functions are also part of the OPERABILITY of the Average Power Range Monitor—Downscale Function (i.e., if either of these IRM Functions cannot send a signal to the Average Power Range Monitor—Downscale Function, the associated Average Power Range Monitor—Downscale channel is considered inoperable).~~

(continued)

Revision J

Db1 INSERT Function 4

The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 12).

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Drywell Pressure—High (continued)

the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

DB7

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and indicative of a LOCA inside primary containment.

Four channels of Drywell Pressure—High Function, 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. The Function is required in MODES 1 and 2 where considerable energy exists in the RCS, resulting in the limiting transients and accidents.

PA3

7a, 7b. Scram Discharge Volume Water Level—High

DB6 either SDV

The SDVs, east and west, are independent with separate drain lines and isolation valves. Each SDV accomodates approximately half of

~~The SDV receives~~ the water displaced by the motion of the CRD pistons during a reactor scram. Should ~~this volume~~ fill to a point where there is insufficient volume to accept the displaced water, control rod insertion would be hindered. Therefore, a reactor scram is initiated while the remaining free volume ~~is~~ still sufficient to accommodate the water from a full core scram. The two types of Scram Discharge Volume Water Level—High Functions are an input to the RPS logic. No credit is taken for a scram initiated from these Functions for any of the design basis accidents or transients analyzed in the FSAR. However, they are retained to ensure the RPS remains OPERABLE.

s are

PA1

PA1 → U (instrument volume portions of the SDVs)

SDV water level is measured by two diverse methods. The level in each of the two SDVs is measured by two float type level switches and two ~~thermal probes~~ for a total of eight level signals. The outputs of these devices are arranged so that there ~~is a signal from a~~ level switch ~~and a thermal probe~~ to each RPS ~~logic~~ channel. The level measurement instrumentation satisfies the recommendations of Reference 8.

differential pressure transmitters

PA1 trip

1b

are either two

signals or two differential pressure transmitter signals

X1 DB3

Each trip channel receives signals from instrumentation from both the east and west SDVs and each RPS trip system receives signals from the two diverse methods.

1B 1B 1B 1B 1B

(continued)

Revision J

PA3

BASES

J

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7a, 7b. Scram Discharge Volume Water Level—High (continued)

DB1

The Allowable Value is chosen low enough to ensure that there is sufficient volume in ~~the~~ each SDV to accommodate the water directed to it from a full scram.

J

Four channels of each type of Scram Discharge Volume Water Level—High Function, with two channels of each type in each trip system, are required to be OPERABLE to ensure that no single ~~instrument~~ failure will preclude a scram from these Functions on a valid signal. These Functions are required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn. At all other times, this Function may be bypassed.

PA2

PA3

DB7

8. Turbine Stop Valve—Closure

Closure of the TSVs results in the loss of ~~a~~ the heat sink ~~that~~ and produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of the transients that would result from the closure of these valves. The Turbine Stop Valve—Closure Function is the primary scram signal for the turbine trips event ~~analyzed in Reference 7~~ (Ref. 10) and feedwater controller failure maximum demand (Ref. 17). For these events, the reactor scram reduces the amount of energy required to be absorbed and ~~, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System,~~ ensures that the MCPR SL is not exceeded.

J

DB7

DB1

DB7

else

Turbine Stop Valve—Closure signals are initiated from position switches located on each of the four TSVs. ~~Two independent~~ One double pole (contact) position switch ~~es are~~ is associated with each stop valve. One of the two ~~switches~~ contacts provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve—Closure channels, each consisting of one position ~~switch~~ contact inputting to a relay. The relay contacts provide a parallel logic input to an RPS trip channel. The logic for the Turbine Stop Valve—Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER ≥ ~~30%~~ 29 % RTP as measured by turbine first stage pressure. This is ~~normally~~ accomplished automatically by pressure transmitters sensing turbine first

DB1

DB1

PA2

(continued)

Revision J

BASES

EHC PA1 PA2

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, ~~Trip~~ Oil Pressure—Low (continued)

(except during required testing or upon actual demand)

automatically by pressure transmitters sensing turbine first stage pressure; therefore, to consider this Function OPERABLE, the turbine bypass valves must remain shut at THERMAL POWER ≥ ~~30%~~ RTP.

DB1 29 DB8

INSERT FUNCTION 9

EHC PA1

The Turbine Control Valve Fast Closure, ~~Trip~~ Oil Pressure—Low Allowable Value is selected high enough to detect imminent TCV fast closure.

and low enough to avoid inadvertent scrams

DB7

EHC PA1

Four channels of Turbine Control Valve Fast Closure, ~~Trip~~ Oil Pressure—Low Function 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. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP, since the Reactor ~~Vessel Steam Dome~~ Pressure—High and the Average Power Range Monitor (Fixed) Neutron Flux—High Functions are adequate to maintain the necessary safety margins.

PA2 29 DB7 PA1

PA1 trip

10. Reactor Mode Switch—Shutdown Position

PA1 trip

The Reactor Mode Switch—Shutdown Position Function provides signals, via the manual scram logic channels, ~~to each of the four RPS logic channels, which~~ are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

directly to the scram pilot valve solenoid power circuits. The manual scram

DB1 DB1 J

The reactor mode switch is a ~~single switch with four channels, each of which provides input into one of the RPS logic channels.~~

DB1

INSERT FUNCTION 10

DB1

There is no Allowable Value for this Function, since the channels are mechanically actuated based solely on reactor mode switch position.

(continued)

Revision J

TA 2

INSERT SR 3.3.1.1.3

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with the applicable extensions. 1J

CLB 2

INSERT SR 3.3.1.1.4

A functional test of each automatic scram contactor is performed to ensure that each automatic RPS trip channel will perform the intended function. There are four RPS channel test switches, one associated with each of the four automatic trip channels (A1, A2, B1, and B2). These test switches allow the operator to test the OPERABILITY of the individual trip channel automatic scram contactors as an alternative to using an automatic scram function trip. This is accomplished by placing the RPS channel test switch in the test position, which will input a trip signal into the associated RPS trip channel. The RPS channel test switches are not specifically credited in the accident analysis. The Manual Scram Functions at JAFNPP are not configured the same as the generic model used in Reference 18. However, Reference 18 concluded that the Surveillance Frequency extensions for RPS Functions were not affected by the difference in configuration since each automatic RPS trip channel has a test switch which is functionally the same as the manual scram switches in the generic model. As such, a functional test of each RPS automatic scram contactor using either its associated test switch or by test of any of the associated automatic RPS Functions is required to be performed once every 7 days. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. In accordance with Reference 18, the scram contactors must be tested as part of the Manual Scram Function. The Frequency of 7 days is based on the reliability analysis of Reference 18. (This automatic scram contactor testing is credited in the analysis to extend many automatic scram Function Surveillance Frequencies).

TA 2

DBG

CLB5

BASES

SURVEILLANCE REQUIREMENTS

4

SR 3.3.1.1.~~5~~ (continued)

CLB2

~~Frequency and is based on the reliability analysis of Reference 10. (The Manual Scram Function's CHANNEL FUNCTIONAL TEST Frequency was credited in the analysis to extend many automatic scram Functions' Frequencies.)~~

5 6 CLB5

SR 3.3.1.1.~~6~~ and SR 3.3.1.1.~~7~~

These Surveillances are established to ensure that no gaps in neutron flux indication exist from subcritical to power operation for monitoring core reactivity status.

The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a neutron flux region without adequate indication. This is required prior to fully withdrawing SRMs ~~from the fully inserted position~~ since indication is being transitioned from the SRMs to the IRMs. OB10 J

The overlap between IRMs and APRMs is of concern when reducing power into the IRM range. On power increases, the system design will prevent further increases (by initiating a rod block) if adequate overlap is not maintained. Overlap between IRMs and APRMs exists when sufficient IRMs and APRMs concurrently have onscale readings such that the transition between MODE 1 and MODE 2 can be made without either APRM downscale rod block, or IRM upscale rod block. Overlap between SRMs and IRMs similarly exists when, prior to fully withdrawing the SRMs ~~from the fully inserted position~~, IRMs are above mid-scale on range 1 before SRMs have reached the upscale rod block. J OB10

6 CLB5

As noted, SR 3.3.1.1.~~7~~ is only required to be met during entry into MODE 2 from MODE 1. That is, after the overlap requirement has been met and indication has transitioned to the IRMs, maintaining overlap is not required (APRMs may be reading downscale once in MODE 2).

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channels that are required in the current MODE or condition should be declared inoperable.

(continued)

Revision J

TA2

INSERT SR 3.3.1.1.8a

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with the applicable extensions. J

CLA6

INSERT SR 3.3.1.1.8

For Function 7.b, the CHANNEL FUNCTIONAL TEST is performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected. For Function 10, the CHANNEL FUNCTIONAL TEST is performed by actually placing the reactor mode switch in the shutdown position. J J

DB9

Move to Pg. B.3.3-29 as indicated

PA2

BASES

Insert SR 3.3.1.1.10

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.10 (continued)

readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

CLB8 CLB8 J

184

The Frequency of 92 days is based on the reliability analysis ~~of Reference 9~~, accuracy, and lower failure rates of the solid-state electronic Analog Transmitter/Trip System components

9

DB9 SR 3.3.1.1.~~11~~ and SR 3.3.1.1.~~13~~ 12

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

INSERT SR 3.3.1.1-12-1

CLB6 PA2

SR 3.3.1.1.9 has been modified by three Notes.

PA2

Note 1 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.~~8~~ 7). 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.

CLB5

CLB9

INSERT SR 3.3.1.1-12-2

DB9

INSERT SR 3.3.1.1.9

~~The Frequency of SR 3.3.1.1.11 is based upon the assumption of a 184 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~~ 12 is based upon the assumption of ~~an~~ a ~~18~~ 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

24 DB8 12 DB9

(continued)

Revision J

PA 2

INSERT SR 3.3.1.1.10

For Functions 8 and 9, this SR is associated with the enabling circuit sensing first stage turbine pressure.

PA 2 CLB6

INSERT SR 3.3.1.1.12-1

For Function 7.b, the CHANNEL CALIBRATION must be performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected. For Functions 8 and 9, SR 3.3.1.1.12 is associated with the enabling circuit sensing first stage turbine pressure as well as the trip function.

CLA9

INSERT SR 3.3.1.1.12-2

Note 3 to SR 3.3.1.1.9 and the Note to SR 3.3.1.1.12 concerns the Neutron Flux - High (Flow Biased) Function (Function 2). Note 3 to SR 3.3.1.1.9 excludes the recirculation loop flow signal portion of the channel, since this portion of the channel is calibrated by SR 3.3.1.1.12. Similarly, the Note to SR 3.3.1.1.12 excludes all portions of the channel except the recirculation loop flow signal portion, since they are covered by SR 3.3.1.1.9.

Reactor Pressure - High and Reactor Vessel Water Level - Low (Level 3) Function sensors (Functions 3 and 4, respectively) are excluded from the RPS RESPONSE TIME testing (Ref. 19). However, prior to the CHANNEL CALIBRATION of these sensors a response check must be performed to ensure adequate response. This testing is required by Reference 20. Personnel involved in this testing must have been trained in response to Reference 21 to ensure they are aware of the consequences of instrument response time degradation. This response check must be performed by placing a fast ramp or a step change into the input of each required sensor. The personnel, must monitor the input and output of the associated sensor so that simultaneous monitoring and verification may be accomplished.

DB9

INSERT SR 3.3.1.1.9

The Frequency of SR 3.3.1.1.9 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

BASES

SURVEILLANCE REQUIREMENTS

DB3

14

SR 3.3.1.1.16 (continued)

during an in-service calibration

bypass valves must remain closed at THERMAL POWER ≥ 30% RTP to ensure that the calibration remains valid.

15 29

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 30% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

EHC

24

The Frequency of 18 months is based on engineering judgment and reliability of the components. X2

15 DB3

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 10. 22

TA1 X1 DB3

PA2

Note 1 excludes

As noted, neutron detectors are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time.

CLB3

INSERT SR 3.3.1.1.15-1

TA1 DB6

CLB4

CLB3 3 2 24

RPS RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Note 2 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 18 month Frequency is consistent with the typical industry 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.

CLB4

CLB4 24

PA2

15

INSERT SR 3.3.1.1.15-2

CLB4

(continued)

Revision J

TA1 CLA3 INSERT SR 3.3.1.1.15-1

since 19 DB6

RPS RESPONSE TIME may be verified by actual response time measurements in any series of sequential, overlapping, or total channel measurements. However, the sensors for Functions 3 and 4 are ~~allowed to be~~ excluded from specific RPS RESPONSE TIME measurement ~~if~~ the conditions of Reference ~~11~~ are satisfied. ~~If these conditions are satisfied~~, sensor response time may be allocated based on either assumed design sensor response time or the manufacturer's stated design response time. ~~When the requirements of Reference 11 are not satisfied~~, sensor response time must be measured. ~~Furthermore, measurement of the~~ instrument loops response times for Functions 3 and 4 is not required if the conditions of Reference 12 are satisfied.

For Functions 3 and 4

For all other Functions

J J J

CLA4 INSERT SR 3.3.1.1.15-2

This ensures all required channels are tested during two Surveillance Frequency intervals. For Functions 2.b, 2.c, 3, 4, 6, and 9, two channels must be tested during each test interval; while for Functions 5 and 8, eight and four channels must be tested, respectively.

BASES (continued) PA1

REFERENCES

1. FSAR, ~~Figure [ ]~~. Section 7.2 DB2
2. FSAR, Section ~~[15.1.2]~~. 14.5.4.2
3. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978. PA6

4. 10 CFR 50.36 (c)(2)(ii) X1

7. ~~4.~~ FSAR, Section ~~[5.2.2]~~. 14.5.1.2 DB2
8. ~~5.~~ FSAR, Section ~~[15.1.38]~~. 14.6.1.2 INSERT REF-1 DB7
11. ~~6.~~ FSAR, Section ~~[6.3.3]~~.
~~7. FSAR, Chapter [15].~~ Insert REF-2 DB7

DB3, X1 renumbering

16. ~~8.~~ P. Check (NRC) letter to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980. PA6
18. ~~9.~~ NEDO-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988. PA2 1 J
22. ~~10.~~ FSAR, Table ~~[7.2-2]~~. 5 DB2 Insert REF-3 CLB3

17. FSAR, Section 14.5.9. 1 J

5. NEDO-31960-A, "BWR Owners' Group Long Term Stability Solutions Licensing Methodology," June 1991.

6. NEDO-31960-A, "BWR Owners' Group Long Term Stability Solutions Licensing Methodology," Supplement 1, March 1992. DB3 1 J

Revision J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.3.1.1 - REACTOR PROTECTION SYSTEM (RPS) INSTRUMENTATION

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB5 The description of the setpoint calculation methodology has been revised to reflect the plant specific methodology.

DB6 The Bases has been revised to reflect the appropriate references.

DB7 The Bases has been revised to reflect the safety analysis. At low powers (e.g., < 29% RTP) the scram from the TSV and TCV is not required; however, the turbine generator can remain online (and trip with resultant pressure transient) below this power level. The TSV and TCV Fast Closure (turbine trip or main generator trip) provide a direct reactor scram when ≥ 29% RTP. When < 29% RTP, a turbine or main generator trip will not result in a direct scram, but should the pressure transient reach the setpoint for the Reactor High Pressure trip, a scram would occur (i.e., is credited to occur from the Reactor High Pressure trip). Since turbine operation below 29% RTP includes MODE 1 and MODE 2, the necessary applicability of the Reactor High Pressure trip is consistent with specifying MODE 1 and 2. References have been included as applicable. Subsequent references have been renumbered as required.

DB8 The Bases has been revised to reflect the setpoint calculation methodology assumptions.

DB9 SR 3.3.1.1.9 has been added to perform a CHANNEL CALIBRATION every 92 days for Function 7.a (Scram Discharge Volume Water Level - High, Differential Pressure Transmitter/Trip Unit) consistent with CTS Table 4.1-2. The Frequency is consistent with the setpoint calculation methodology for this Function. In addition, the Frequency for ISTS SR 3.3.1.1.11, the 184 day CHANNEL CALIBRATION requirement for the APRM Functions, has been changed to 92 days (ITS SR 3.3.1.1.9), consistent with the CTS. The Bases description has been reordered and renumbered as required.

DB10 Changes have been made to reflect those changes made to the Specification.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 332, Revision 1 have been incorporated into the revised Improved Technical Specifications. However, NEDO-32291-A, Supplement 1 has not yet been adopted by JAFNPP. Therefore, this portion of the TSTF has not been incorporated.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.3.1.1 - REACTOR PROTECTION SYSTEM (RPS) INSTRUMENTATION

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA2 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 205, Revision 3 have been incorporated into the revised Improved Technical Specifications.

TA3 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 231, Revision 1 have been incorporated into the revised Improved Technical Specifications.

TA4 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 355, Revision 0, as modified by WOG-ED-25, have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 NUREG-1433, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995. Subsequent References have been renumbered, as applicable.

X2 The SR 3.3.1.1.13 and SR 3.3.1.1.14 Frequencies have been modified from 18 months to 24 months consistent with the JAFNPP fuel cycle.

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTES-----------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	-----------------NOTE----------------- Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. --- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint," while operating at ≥ 25% RTP.	7 days
SR 3.3.1.1.3	-----------------NOTE----------------- Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.4	Perform a functional test of each RPS automatic scram contactor.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to fully withdrawing SRMs
SR 3.3.1.1.6	------------------NOTE------------------ Only required to be met during entry into MODE 2 from MODE 1. -- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1000 MWD/T average core exposure
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	-----------------NOTES----------------- 1. Neutron detectors are excluded. 2. For Functions 1.a and 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 recirculation loop flow signal portion of the channel is excluded. -- Perform CHANNEL CALIBRATION.	92 days

(continued)

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 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.13	≤ 120/125 divisions of full scale
	5[(a)]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.13	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.13	NA
	5[(a)]	3	H	SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.13	NA
2. Average Power Range Monitors					
a. Neutron Flux – High, (Startup)	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.13	≤ 15% RTP
b. Neutron Flux-High (Flow Biased)	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.15	As specified in the COLR and ≤ 117% RTP

(continued)

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

Table 3.3.1.1-1 (page 3 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
7. Scram Discharge Volume Water Level – High					
a. Differential Pressure Transmitter/Trip Unit	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.13	≤ 34.5 gallons
	5(a)	2	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.13	≤ 34.5 gallons
b. Level Switch	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.13	≤ 34.5 gallons
	5(a)	2	H	SR 3.3.1.1.4 SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.13	≤ 34.5 gallons
8. Turbine Stop Valve – Closure	≥ 29% RTP	4	E	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 15% closed
9. Turbine Control Valve Fast Closure, EHC Oil Pressure – Low	≥ 29% RTP	2	E	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 500 psig and ≤ 850 psig
10. Reactor Mode Switch – Shutdown Position	1,2	1	G	SR 3.3.1.1.11 SR 3.3.1.1.13	NA
	5(a)	1	H	SR 3.3.1.1.11 SR 3.3.1.1.13	NA
11. Manual Scram	1,2	1	G	SR 3.3.1.1.8 SR 3.3.1.1.13	NA
	5(a)	1	H	SR 3.3.1.1.8 SR 3.3.1.1.13	NA

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

BASES

BACKGROUND (continued)

Technical Specifications in order to define OPERABILITY of the devices and is designated as the Allowable Value which, as stated above, is the same as the LSSS.

The Allowable Values specified in Table 3.3.1.1-1 serve as the LSSS such that a channel is OPERABLE if the trip setpoint is found not to exceed the Allowable Value. As such, the Allowable Value differs from the Trip Setpoint by an amount primarily equal to the expected instrument loop uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the device will still meet the LSSS definition and ensure that a Safety Limit is not exceeded at any given point of time as long as the device has not drifted beyond that expected during the surveillance interval. If the actual setting of the device is found to have exceeded the Allowable Value the device would be considered inoperable from a Technical Specification perspective. This requires corrective action including those actions required by 10 CFR 50.36 when automatic protective devices do not function as required. Note that, although the channel is "OPERABLE" under these circumstances, the trip setpoint should be left adjusted to a value within the established trip setpoint calibration tolerance band, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowance of the uncertainty terms assigned.

The RPS, as described in the UFSAR, Section 7.2 (Ref. 1), includes sensors, relays, logic circuits, bypass circuits, and switches that are necessary to cause initiation of a reactor scram. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. The input parameters to the scram logic are from instrumentation that monitors reactor vessel water level, reactor vessel pressure, neutron flux, main steam line isolation valve position, turbine control valve (TCV) fast closure, EHC Oil Pressure-Low, turbine stop valve (TSV) position, drywell pressure, and scram discharge volume (SDV) water level, as well as reactor mode switch in shutdown position and manual scram signals. There are at least four redundant sensor input signals from each of these parameters (with the exception of the reactor mode switch in shutdown position and manual scram signals). Most channels include instrumentation that compares measured input signals with pre-established setpoints. When the setpoint is exceeded,

(continued)

BASES

BACKGROUND (continued)

the channel outputs an RPS trip signal to the trip logic.

The RPS is comprised of two independent trip systems (A and B) with three trip channels in each trip system (trip channels A1, A2, and A3, B1, B2, and B3) as described in Reference 1. Trip channels A1, A2, B1, and B2 contain automatic protective instrument logic. The above monitored parameters are represented by at least one input to each of these automatic trip channels. The outputs of the automatic trip channels in a trip system are combined in a one-out-of-two logic so that either channel can trip the associated trip system. The tripping of both trip systems will produce a reactor scram. This logic arrangement is referred to as a one-out-of-two taken twice logic. There are four RPS channel test switches, one associated with each of the four automatic trip channels. These test switches allow the operator to test the OPERABILITY of the individual trip channel automatic scram contactors. In addition, trip channels A3 and B3 (one trip channel per trip system) are provided for manual scram. Placing the reactor mode switch in shutdown position or depressing both channel push buttons (one per trip system) will initiate the manual trip function. Each trip system is reset by use of a reset switch. If a full scram occurs (both trip systems trip), a relay prevents reset of the trip systems for approximately 10 seconds after the full scram signal is received. This 10 second delay on reset ensures that the scram function will be completed.

Two scram pilot valves are located in the hydraulic control unit for each control rod drive (CRD). Each scram pilot valve is solenoid operated, with the solenoids normally energized. The scram pilot valves control the air supply to the scram inlet and outlet valves for the associated CRD. When either scram pilot valve solenoid is energized, air pressure holds the scram valves closed and, therefore, both scram pilot valve solenoids must be de-energized to cause a control rod to scram. The scram valves control the supply and discharge paths for the CRD water during a scram. One of the scram pilot valve solenoids for each CRD is controlled by trip system A, and the other solenoid is controlled by trip system B. Any trip of trip system A in conjunction with any trip in trip system B results in de-energizing both solenoids, air bleeding off, scram valves opening, and control rod scram.

(continued)

BASES

BACKGROUND (continued)

The backup scram valves, which energize on a scram signal to depressurize the scram air header, are also controlled by the RPS. Additionally, the RPS System controls the SDV vent and drain valves such that when both trip systems trip, the SDV vent and drain valves close to isolate the SDV.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The actions of the RPS are assumed in the safety analyses of References 1, 2, and 3. The RPS is required to initiate a reactor scram when monitored parameter values exceed the Allowable Values, specified by the setpoint methodology and listed in Table 3.3.1.1-1 to preserve the integrity of the fuel cladding, the RCPB, and the containment by minimizing the energy that must be absorbed following a LOCA.

RPS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4). Functions not specifically credited in the accident analysis are retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The OPERABILITY of the RPS is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.1.1-1. Each Function must have a required number of OPERABLE channels per RPS trip system, with their setpoints within the specified Allowable Value, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Each channel must also respond within its assumed response time, where appropriate.

Allowable Values are specified, as appropriate, for RPS Functions specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Intermediate Range Monitor (IRM)

1.a. Intermediate Range Monitor Neutron Flux - High

The IRMs monitor neutron flux levels from the upper range of the source range monitor (SRM) to the lower range of the average power range monitors (APRMs). The IRMs are capable of generating trip signals that can be used to prevent fuel damage resulting from abnormal operating transients in the intermediate power range. In this power range, the most significant source of reactivity change is due to control rod withdrawal. The IRM provides diverse protection for the rod worth minimizer (RWM), which monitors and controls the movement of control rods at low power. The RWM prevents the withdrawal of an out of sequence control rod during startup that could result in an unacceptable neutron flux excursion (Ref. 2). The IRM provides mitigation of the neutron flux excursion. To demonstrate the capability of the IRM System to mitigate control rod withdrawal events, a generic analysis has been performed (Ref. 3) to evaluate the consequences of control rod withdrawal events during startup that are mitigated only by the IRM. This analysis, which assumes that one IRM channel in each trip system is bypassed, demonstrates that the IRMs provide protection against local control rod withdrawal errors and results in peak fuel enthalpy below the 170 cal/gm fuel failure threshold criterion.

The IRMs are also capable of limiting other reactivity excursions during startup, such as cold water injection events, although no credit is specifically assumed.

The IRM System is divided into two groups of IRM channels, with four IRM channels inputting to each trip system. The analysis of Reference 3 assumes that one channel in each trip system is bypassed. Therefore, six channels with three channels in each trip system are required for IRM OPERABILITY to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This trip is active in each of the 10 ranges of the IRM, which must be selected by the operator to maintain the neutron flux within the monitored level of an IRM range.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

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 Neutron Flux - High (Fixed) Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events. The APRM Neutron Flux - High (Startup) Function is bypassed when the reactor mode switch is in the run position.

2.b. Average Power Range Monitor Neutron Flux - High (Flow Biased)

The Average Power Range Monitor Neutron Flux - High (Flow Biased) Function monitors neutron flux and approximates the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux trip level is varied as a function of recirculation drive flow but is clamped at an upper limit that is lower than the Average Power Range Monitor Neutron Flux - High (Fixed) Function, Function 2.c, Allowable Value. The Average Power Range Monitor Neutron Flux - High (Flow Biased) Function provides protection against transients where THERMAL POWER increases slowly (such as the loss of feedwater heating event), however, no credit is taken for this Function in the safety analyses except in the case of the thermal-hydraulic instability analysis. This protection is primarily achieved by the clamped portion of the Allowable Value. The APRM Neutron Flux - High (Flow Biased) Function will suppress power oscillations prior to exceeding the fuel safety limit (MCPR) caused by thermal hydraulic instability. As described in References 5 and 6, this protection is provided at a high statistical confidence level for core-wide mode oscillations and at a nominal statistical confidence level for regional mode oscillations. References 5 and 6 also show that the core-wide mode

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Neutron Flux-High (Flow Biased) (continued)

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 Neutron Flux-High (Flow Biased) channels in the associated trip system must be considered inoperable).

The flow biased Allowable Value is credited in the safety analyses (thermal- hydraulic instability) and is specifically confirmed for each operating cycle. For this reason the Allowable Value is included in the COLR for both single and two recirculation loop operation. The clamped portion of the Allowable Value is set more conservative than the APRM Neutron Flux-High (Fixed) (Function 2.c). J

The Average Power Range Monitor Neutron Flux-High (Flow Biased) Function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive THERMAL POWER and potentially exceeding the SL applicable to high pressure and core flow conditions (MCPR SL). During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

2.c. Average Power Range Monitor Neutron Flux-High (Fixed)

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 Neutron Flux-High (Fixed) Function is capable of generating a trip signal to prevent fuel damage or excessive Reactor Coolant System (RCS) pressure. For the overpressurization protection analysis of Reference 7, the Average Power Range Monitor Neutron Flux-High (Fixed) Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 8) takes credit for the Average Power Range Monitor Neutron Flux-High (Fixed) Function to terminate the CRDA.

The APRM System is divided into two groups of channels with three APRM channels providing input to each trip system.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.c. Average Power Range Monitor Neutron Flux – High (Fixed) (continued)

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 Neutron Flux – High (Fixed) 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 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.

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor Neutron Flux – High (Fixed) Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Neutron Flux – High (Fixed) Function is assumed in the CRDA analysis (Ref. 8), which is applicable in MODE 2, the Average Power Range Monitor Neutron Flux – High (Startup) Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Range Monitor Neutron Flux – High (Fixed) Function is not required in MODE 2.

2.d. Average Power Range Monitor – Inop

This signal provides assurance that a minimum number of APRMs are OPERABLE. Anytime an APRM Operate-Calibrate switch is moved to any position other than "Operate," an APRM module is unplugged, or the APRM has too few LPRM inputs (< 11), an inoperative trip signal will be received by the RPS, unless the APRM is bypassed. Since only one APRM in each trip system may be bypassed, only one APRM in each trip system may be inoperable without resulting in an RPS trip signal. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Reactor Pressure - High (continued)

in MODES 1 and 2 when the RCS is pressurized and the potential for pressure increase exists.

4. Reactor Vessel Water Level - Low (Level 3)

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, a reactor scram is initiated at Level 3 to substantially reduce the heat generated in the fuel from fission. The Reactor Vessel Water Level - Low (Level 3) Function is one of the Functions assumed in the analysis of the recirculation line break (Ref. 11). The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the Emergency Core Cooling Systems (ECCS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level - Low (Level 3) signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

Four channels of Reactor Vessel Water Level - Low (Level 3) Function, 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 failure will preclude a scram from this Function on a valid signal.

The Reactor Vessel Water Level - Low (Level 3) Allowable Value is selected to ensure that during normal operation the separator skirts are not uncovered (this protects available recirculation pump net positive suction head (NPSH) from significant carryunder) and, for transients involving loss of all normal feedwater flow, initiation of the low pressure ECCS subsystems at Reactor Vessel Water - Low Low Low (Level 1) will not be required. The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 12). J

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7.a, 7.b. Scram Discharge Volume Water Level - High

The SDVs, east and west, are independent with separate drain lines and isolation valves. Each SDV accommodates approximately half of the water displaced by the motion of the CRD pistons during a reactor scram. Should either SDV fill to a point where there is insufficient volume to accept the displaced water, control rod insertion would be hindered. Therefore, a reactor scram is initiated while the remaining free volumes are still sufficient to accommodate the water from a full core scram. The two types of Scram Discharge Volume Water Level - High Functions are an input to the RPS logic. No credit is taken for a scram initiated from these Functions for any of the design basis accidents or transients analyzed in the UFSAR. However, they are retained to ensure the RPS remains OPERABLE.

SDV water level is measured by two diverse methods. The level in each of the two SDVs (instrument volume portions of the SDVs) is measured by two float type level switches and two differential pressure transmitters for a total of eight level signals. The outputs of these devices are arranged so that there are either two level switch signals or two differential pressure transmitter signals to each RPS trip channel. Each trip channel receives signals from instrumentation from both the east and west SDVs and each RPS trip system receives signals from the two diverse methods. The level measurement instrumentation satisfies the recommendations of Reference 16.

The Allowable Value is chosen low enough to ensure that there is sufficient volume in each SDV to accommodate the water directed to it from a full scram.

Four channels of each type of Scram Discharge Volume Water Level - High Function, with two channels of each type in each trip system, are required to be OPERABLE to ensure that no single failure will preclude a scram from these Functions on a valid signal. These Functions are required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn. At all other times, this Function may be bypassed.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

8. Turbine Stop Valve - Closure

Closure of the TSVs results in the loss of the heat sink and produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of the transients that would result from the closure of these valves. The Turbine Stop Valve - Closure Function is the primary scram signal for the turbine trip (Ref. 10) and feedwater controller failure - maximum demand (Ref. 17) events. For these events, the reactor scram reduces the amount of energy required to be absorbed and ensures that the MCPR SL is not exceeded. J

Turbine Stop Valve - Closure signals are initiated from position switches located on each of the four TSVs. One double pole (contact) position switch is associated with each stop valve. One of the two contacts provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve - Closure channels, each consisting of one position switch contact inputting to a relay. The relay contacts provide a parallel logic input to an RPS trip channel. The logic for the Turbine Stop Valve - Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER ≥ 29% RTP as measured by turbine first stage pressure. This is accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, to consider this Function OPERABLE, the turbine bypass valves must remain shut (except during required testing or upon actual demand) at THERMAL POWER ≥ 29% RTP. In addition, other steam loads, such as second stage reheaters in operation, must be accounted for in establishing the setpoint for turbine first stage pressure. Otherwise, the setpoint would be non-conservative with respect to the 29% RTP RPS bypass.

The Turbine Stop Valve - Closure Allowable Value is selected to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve - Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single failure will preclude

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, EHC Oil Pressure - Low (continued)

detect imminent TCV fast closure and low enough to avoid inadvertent scrams.

Four channels of Turbine Control Valve Fast Closure, EHC Oil Pressure - Low Function 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 failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 29% RTP. This Function is not required when THERMAL POWER is < 29% RTP, since the Reactor Pressure - High and the Average Power Range Monitor Neutron Flux - High (Fixed) Functions are adequate to maintain the necessary safety margins.

10. Reactor Mode Switch - Shutdown Position

The Reactor Mode Switch - Shutdown Position Function provides signals, via the manual scram trip channels, directly to the scram pilot valve solenoid power circuits. The manual scram trip channels are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a keylock four-position, four-bank switch. The reactor mode switch will scram the reactor if it is placed in the shutdown position. Scram signals from the reactor mode switch are input into each of the two RPS manual scram trip channels.

There is no Allowable Value for this Function, since the channels are mechanically actuated based solely on reactor mode switch position.

Two channels of Reactor Mode Switch - Shutdown Position Function, with one channel in each trip system, are available and required to be OPERABLE. The Reactor Mode Switch - Shutdown Position Function is required to be OPERABLE in MODES 1 and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.2 (continued)

MFLPD. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.8.

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.3

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with the applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. | J

As noted, SR 3.3.1.1.3 is not required to be performed when entering MODE 2 from MODE 1, since testing of the MODE 2 required IRM and APRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.3 (continued)

consideration of providing a reasonable time in which to complete the SR.

A Frequency of 7 days provides an acceptable level of system average unavailability over the Frequency interval and is based on reliability analysis (Ref. 18).

SR 3.3.1.1.4

A functional test of each automatic scram contactor is performed to ensure that each automatic RPS trip channel will perform the intended function. There are four RPS channel test switches, one associated with each of the four automatic trip channels (A1, A2, B1, and B2). These test switches allow the operator to test the OPERABILITY of the individual trip channel automatic scram contactors as an alternative to using an automatic scram function trip. This is accomplished by placing the RPS channel test switch in the test position, which will input a trip signal into the associated RPS trip channel. The RPS channel test switches are not specifically credited in the accident analysis. The Manual Scram Functions at JAFNPP are not configured the same as the generic model used in Reference 18. However, Reference 18 concluded that the Surveillance Frequency extensions for RPS Functions were not affected by the difference in configuration since each automatic RPS trip channel has a test switch which is functionally the same as the manual scram switches in the generic model. As such, a functional test of each RPS automatic scram contactor using either its associated test switch or by test of any of the associated automatic RPS Functions is required to be performed once every 7 days. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. In accordance with Reference 18, the scram contactors must be tested as part of the Manual Scram Function. The Frequency of 7 days is based on the reliability analysis of Reference 18. (This automatic scram contactor testing is credited in the analysis to extend many automatic Scram Function Surveillance Frequencies). J

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.5 and SR 3.3.1.1.6

These Surveillances are established to ensure that no gaps in neutron flux indication exist from subcritical to power operation for monitoring core reactivity status.

The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a neutron flux region without adequate indication. This is required prior to fully withdrawing SRMs since indication is being transitioned from the SRMs to the IRMs.

The overlap between IRMs and APRMs is of concern when reducing power into the IRM range. On power increases, the system design will prevent further increases (by initiating a rod block) if adequate overlap is not maintained. Overlap between IRMs and APRMs exists when sufficient IRMs and APRMs concurrently have onscale readings such that the transition between MODE 1 and MODE 2 can be made without either APRM downscale rod block, or IRM upscale rod block. Overlap between SRMs and IRMs similarly exists when, prior to fully withdrawing the SRMs, IRMs are above mid-scale on range 1 before SRMs have reached the upscale rod block.

As noted, SR 3.3.1.1.6 is only required to be met during entry into MODE 2 from MODE 1. That is, after the overlap requirement has been met and indication has transitioned to the IRMs, maintaining overlap is not required (APRMs may be reading downscale once in MODE 2).

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channels that are required in the current MODE or condition should be declared inoperable.

A Frequency of 7 days is reasonable based on engineering judgment and the reliability of the IRMs and APRMs.

SR 3.3.1.1.7

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP)

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SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.7 (continued)

System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1000 MWD/T Frequency is based on operating experience with LPRM sensitivity changes.

SR 3.3.1.1.8 and SR 3.3.1.1.11

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with the applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. For Function 2.b, the CHANNEL FUNCTIONAL TEST includes the adjustment of the APRM channel to conform to the calibrated flow signal. This ensures that the total loop drive flow signals from the flow units used to vary the setpoint is appropriately compared to a valid core flow signal to verify the flow signal trip setpoint and, therefore, the APRM Function accurately reflects the required setpoint as a function of flow. If the flow unit signal is not within the appropriate flow limit, one required APRM that receives an input from the inoperable flow unit must be declared inoperable. For Function 7.b, the CHANNEL FUNCTIONAL TEST is performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected. For Function 10, the CHANNEL FUNCTIONAL TEST is performed by actually placing the reactor mode switch in the shutdown position.

The 92 day Frequency of SR 3.3.1.1.8 is based on the reliability analysis of Reference 18.

The 24 month Frequency of SR 3.3.1.1.11 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.8 and SR 3.3.1.1.11 (continued)

reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.1.1.9 and SR 3.3.1.1.12

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. For Function 7.b, the CHANNEL CALIBRATION must be performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected. For Functions 8 and 9, SR 3.3.1.1.12 is associated with the enabling circuit sensing first stage turbine pressure as well as the trip function.

SR 3.3.1.1.9 has been modified by three Notes. Note 1 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.7). 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 3.3.1.1.9 and the Note to SR 3.3.1.1.12 concerns the Neutron Flux—High (Flow Biased) Function (Function 2). Note 3 to SR 3.3.1.1.9 excludes the recirculation loop flow signal portion of the channel, since

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.9 and SR 3.3.1.1.12 (continued)

this portion of the channel is calibrated by SR 3.3.1.1.12. Similarly, the Note to SR 3.3.1.1.12 excludes all portions of the channel except the recirculation loop flow signal portion, since they are covered by SR 3.3.1.1.9.

Reactor Pressure – High and Reactor Vessel Water Level – Low (Level 3) Function sensors (Functions 3 and 4, respectively) are excluded from the RPS RESPONSE TIME testing (Ref. 19). However, prior to the CHANNEL CALIBRATION of these sensors a response check must be performed to ensure adequate response. This testing is required by Reference 20. Personnel involved in this testing must have been trained in response to Reference 21 to ensure they are aware of the consequences of instrument response time degradation. This response check must be performed by placing a fast ramp or a step change into the input of each required sensor. The personnel, must monitor the input and output of the associated sensor so that simultaneous monitoring and verification may be accomplished.

The Frequency of SR 3.3.1.1.9 is based on 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 3.3.1.1.12 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 3.3.1.1.10

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.1.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology. For Functions 8 and 9, this SR is associated with the enabling circuit sensing first stage turbine pressure.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.14 (continued)

placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

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

SR 3.3.1.1.15

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The RPS RESPONSE TIME acceptance criteria are included in Reference 22.

RPS RESPONSE TIME may be verified by actual response time measurements in any series of sequential, overlapping, or total channel measurements. However, the sensors for Functions 3 and 4 are excluded from specific RPS RESPONSE TIME measurement since the conditions of Reference 19 are satisfied. For Functions 3 and 4, sensor response time may be allocated based on either assumed design sensor response time or the manufacturer's stated design response time. For all other Functions, sensor response time must be measured.

Note 1 excludes neutron detectors from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time.

RPS RESPONSE TIME tests are conducted on a 24 month STAGGERED TEST BASIS. Note 2 requires STAGGERED TEST BASIS Frequency to be determined based on 2 channels. This ensures all required channels are tested during two Surveillance Frequency intervals. For Functions 2.b, 2.c, 3, 4, 6, and 9, two channels must be tested during each test; while for Functions 5 and 8, eight and four channels must be tested. 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 refueling cycle and is based upon plant operating experience, which shows that random failures of

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.15 (continued)

instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

REFERENCES

1. UFSAR, Section 7.2.
2. UFSAR, Section 14.5.4.2.
3. NEDO-23842, Continuous Control Rod Withdrawal Transient In The Startup Range, April 18, 1978.
4. 10 CFR 50.36(c)(2)(ii).
5. NEDO-31960-A, BWR Owners' Group Long Term Stability Solutions Licensing Methodology, June 1991.
6. NEDO-31960-A, Supplement 1, BWR Owners' Group Long Term Stability Solutions Licensing Methodology, Supplement 1, March 1992.
7. UFSAR, Section 14.5.1.2.
8. UFSAR, Section 14.6.1.2.
9. UFSAR, Section 14.5.2.1.
10. UFSAR, Section 14.5.2.2.
11. UFSAR, Section 6.3.
12. Drawing 11825-5.01-15D, Rev. D, Reactor Assembly Nuclear Boiler, (GE Drawing 919D690BD).
13. UFSAR, Section 14.5.5.1.
14. UFSAR, Section 14.5.2.3.
15. UFSAR, Section 14.6.1.5.
16. P. Check (NRC) letter to G. Lainas (NRC), BWR Scram Discharge System Safety Evaluation, December 1, 1980.

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DISCUSSION OF CHANGES
ITS: 3.3.1.2 - SOURCE RANGE MONITOR (SRM) INSTRUMENTATION

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 The current requirement for SRM response of 3 cps is based upon a signal to noise ratio of ≥ 2:1. This is implicit in CTS 4.3.B.4. Thus, the explicit requirement in ITS SR 3.3.1.2.4 to verify 3.0 cps with a signal to noise ratio ≥ 2:1 is considered an administrative change.

A3 The CTS does not have a specific CHANNEL CALIBRATION requirement for the SRM indication. However, the CTS does have a 92 day CHANNEL CALIBRATION for the MODE 2 SRM Control Rod Block Function. Therefore, consistent with this CTS requirement and with current practice, a Surveillance Requirement is included as ITS SR 3.3.1.2.7 to perform a CHANNEL CALIBRATION every 92 days.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.3.B.4 and 4.3.B.4 require two Source Range Monitors (SRMs) to be Operable whenever control rods are withdrawn for startup or during refueling. ITS LCO 3.3.1.2 (Table 3.3.1.2-1) will require three SRMs to be Operable at all times in MODE 2 prior to and during control rod withdrawal until the flux level is sufficient to maintain the Intermediate Range Monitor (IRM) on Range 3 or above in MODE 2 (Table 3.3.1.2-1 Footnote a). This requirement for an additional SRM to be Operable is more restrictive change and will ensure adequate SRMs are Operable during reactor startup. This is consistent with NUREG-1433, Revision 1.

M2 CTS 3.3.B.4 and 4.3.B.4 require that SRMs be Operable when control rods are withdrawn for startup or during refueling. CTS 3.10.B and 4.10.B require the SRMs to be Operable during "Core Alterations." There are no requirements for SRM Operability during MODE 3 and MODE 4. ITS LCO 3.3.1.2 (Table 3.3.1.2-1) will require 2 SRM channels to be Operable at all times in MODE 3 and MODE 4 because the SRMs are the primary indication of neutron flux levels in these MODES. Additionally, SRM Operability in MODES 3 and 4 must be demonstrated by the performance of ITS SR 3.3.1.2.3 (CHANNEL CHECK), SR 3.3.1.2.4 (count rate verification), SR 3.3.1.2.6, (CHANNEL FUNCTIONAL TEST), and SR 3.3.1.2.7

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
[4.3.B.4/M3/A2] [3.10.B.2] [M2] [M7] SR 3.3.1.2.4	------------------NOTE------------------ Not required to be met with less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant. -- Verify count rate is: a. ≥ [3.0] cps with a signal to noise ratio ≥ [2:1] ~~or~~ b. ≥ [0.7] cps with a signal to noise ratio ≥ [20:1].	12 hours during CORE ALTERATIONS AND 24 hours CLB1
[4.10.B] [M10] SR 3.3.1.2.5	Perform CHANNEL FUNCTIONAL TEST ~~[~~and determination of signal to noise ratio~~]~~.	7 days DB2
[M4] [M2] SR 3.3.1.2.6	------------------NOTE------------------ Not required to be performed until 12 hours after IRMs on Range 2 or below. -- Perform CHANNEL FUNCTIONAL TEST ~~[~~and determination of signal to noise ratio~~]~~.	31 days
[M2] [M9] [A3] SR 3.3.1.2.7	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. Not required to be performed until 12 hours after IRMs on Range 2 or below. -- Perform CHANNEL CALIBRATION.	92 days CLB2 ~~[18] months~~

------------------NOTE------------------
The determination of signal to noise ratio is not required to be met with less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant.
X1

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JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.3.1.2 - SOURCE RANGE MONITOR (SRM) INSTRUMENTATION

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The JAFNPP is not licensed with the option for utilizing a lower count rate. Therefore, this option in ISTS SR 3.3.1.2.4.b has not been used in the JAFNPP ITS. J

CLB2 The bracketed Frequency of 18 months in SR 3.3.1.2.7 has been changed to 92 days consistent with the SRM Control Rod Block Function Channel Calibration Frequency in the CTS.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Typographical/grammatical correction made.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the number of required SRM channels during MODE 2 operations of the three (3) has been included consistent with the values in ITS ACTION B and in Table 3.3.1.2-1 for MODE 2 operations. JAFNPP design is consistent with the Standard and this requirement has been added in accordance with M1.

DB2 The brackets have been removed in ITS SR 3.3.1.2.5 and SR 3.3.1.2.6 and the requirement to perform the determination of the signal to noise ratio along with the CHANNEL FUNCTIONAL TEST maintained since it is an important requirement for SRM OPERABILITY as discussed in the Bases.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 A new Note has been added to ISTS SR 3.3.1.2.5 to state that the determination of the signal to noise ratio is not required to be met with less than or equal to four fuel assemblies adjacent to the SRM and

BASES

ACTIONS

D.1 and D.2 (continued)

control rods ensures that the reactor will be at its minimum reactivity level while no neutron monitoring capability is available. Placing the reactor mode switch in the shutdown position prevents subsequent control rod withdrawal by maintaining a control rod block. The allowed Completion Time of 1 hour is sufficient to accomplish the Required Action, and takes into account the low probability of an event requiring the SRM occurring during this interval.

E.1 and E.2

PA3 PA4 PA3

(S) or nonexistent

With one or more required SRMs inoperable in MODE 5, the ability to detect local reactivity changes in the core during refueling is degraded. CORE ALTERATIONS must be immediately suspended and action must be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Suspending CORE ALTERATIONS prevents the two most probable causes of reactivity changes, fuel loading and control rod withdrawal, from occurring. Inserting all insertable control rods ensures that the reactor will be at its minimum reactivity given that fuel is present in the core. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe, conservative position.

J

Action (once required to be initiated) to insert control rods must continue until all insertable rods in core cells containing one or more fuel assemblies are inserted.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each SRM Applicable MODE or other specified conditions are found in the SRs column of Table 3.3.1.2-1.

PA3

SR 3.3.1.2.1 and SR 3.3.1.2.3

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on another channel. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the

(continued)

Revision J

(TA1) INSERT SR 3.3.1.2.5

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with the applicable extensions. | (5)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2.5 and SR 3.3.1.2.6 (continued)

CHANNEL CHECK) that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

Verification of the signal to noise ratio also ensures that the detectors are inserted to an acceptable operating level. In a fully withdrawn condition, the detectors are sufficiently removed from the fueled region of the core to essentially eliminate neutrons from reaching the detector. Any count rate obtained while the detectors are fully withdrawn is assumed to be "noise" only.

Insert SR

PA1

SR 3.3.1.2.6

The Note to ~~the Surveillance~~ allows the Surveillance to be delayed until entry into the specified condition of the Applicability (THERMAL POWER decreased to IRM Range 2 or below). The SR must be performed within 12 hours after IRMs are on Range 2 or below. The allowance to enter the Applicability with the 31 day Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. ~~In this event, the~~ Twelve ~~12~~ hour ~~Frequency~~ is reasonable, based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances.

PA3

PA3

J

CLB1

SR 3.3.1.2.7

PA3

monitors

J

Performance of a CHANNEL CALIBRATION at a Frequency of 92 days ~~18 months~~ verifies the performance of the SRM ~~detectors~~ 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 neutron detectors are excluded from the CHANNEL CALIBRATION (Note 1) because they cannot readily be adjusted. The detectors are fission chambers that are designed to have a relatively constant sensitivity over the range and with an accuracy specified for a fixed useful life.

PA3

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2.7 (continued)

Note 2 to the Surveillance allows the Surveillance to be delayed until entry into the specified condition of the Applicability. The SR must be performed in MODE 2 within 12 hours of entering MODE 2 with IRMs on Range 2 or below. The allowance to enter the Applicability with the ~~18 month~~ 92 day Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. ~~In this event,~~ Twelve ~~the 12~~ hours ~~Frequency~~ is reasonable, based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances. CLB1 PA3 T

REFERENCES None.

Revision 5

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.2.4	-----NOTE----- Not required to be met with less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant. ----- Verify count rate is $\geq$ 3.0 cps with a signal to noise ratio $\geq$ 2:1.	12 hours during CORE ALTERATIONS **AND** 24 hours
SR 3.3.1.2.5	-----NOTE----- The determination of signal to noise ratio is not required to be met with less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant. ----- Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio.	7 days
SR 3.3.1.2.6	-----NOTE----- Not required to be performed until 12 hours after IRMs on Range 2 or below. ----- Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio.	31 days

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

With one or more required SRMs inoperable in MODE 3 or 4, the neutron flux monitoring capability is degraded or nonexistent. The requirement to fully insert all insertable control rods ensures that the reactor will be at its minimum reactivity level while no neutron monitoring capability is available. Placing the reactor mode switch in the shutdown position prevents subsequent control rod withdrawal by maintaining a control rod block. The allowed Completion Time of 1 hour is sufficient to accomplish the Required Action, and takes into account the low probability of an event requiring the SRM occurring during this interval.

E.1 and E.2

With one or more required SRMs inoperable in MODE 5, the ability to detect local reactivity changes in the core during refueling is degraded or nonexistent. CORE ALTERATIONS must be immediately suspended and action must be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Suspending CORE ALTERATIONS prevents the two most probable causes of reactivity changes, fuel loading and control rod withdrawal, from occurring. Inserting all insertable control rods ensures that the reactor will be at its minimum reactivity given that fuel is present in the core. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe, conservative position.

Action (once required to be initiated) to insert control rods must continue until all insertable rods in core cells containing one or more fuel assemblies are inserted.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each SRM Applicable MODE or other specified conditions are found in the SRs column of Table 3.3.1.2-1.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2.5 and SR 3.3.1.2.6

Performance of a CHANNEL FUNCTIONAL TEST demonstrates the associated channel will function properly. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with the applicable extensions. SR 3.3.1.2.5 is required in MODE 5, and the 7 day Frequency ensures that the channels are OPERABLE while core reactivity changes could be in progress. This Frequency is reasonable, based on operating experience and on other Surveillances (such as a CHANNEL CHECK), that ensure proper functioning between CHANNEL FUNCTIONAL TESTS. 1(3)

SR 3.3.1.2.6 is required in MODE 2 with IRMs on Range 2 or below, and in MODES 3 and 4. Since core reactivity changes do not normally take place in MODES 3 and 4, and core reactivity changes are due only to control rod movement in MODE 2, the Frequency has been extended from 7 days to 31 days. The 31 day Frequency is based on operating experience and on other Surveillances (such as CHANNEL CHECK) that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

Verification of the signal to noise ratio also ensures that the detectors are inserted to an acceptable operating level. In a fully withdrawn condition, the detectors are sufficiently removed from the fueled region of the core to essentially eliminate neutrons from reaching the detector. Any count rate obtained while the detectors are fully withdrawn is assumed to be "noise" only.

With few fuel assemblies loaded, the SRMs will not have a high enough count rate to determine the signal to noise ratio. Therefore, allowances are made for loading sufficient "source" material, in the form of irradiated fuel assemblies, to establish the conditions necessary to determine the signal to noise ratio. To accomplish this, SR 3.3.1.2.5 is modified by a Note that states that the determination of signal to noise ratio is not required to be

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2.5 and SR 3.3.1.2.6 (continued)

met on an SRM that has less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies are in the associated core quadrant. With four or less fuel assemblies loaded around each SRM and no other fuel assemblies in the associated quadrant, even with a control rod withdrawn the configuration will not be critical.

The Note to SR 3.3.1.2.6 allows the Surveillance to be delayed until entry into the specified condition of the Applicability (THERMAL POWER decreased to IRM Range 2 or below). The SR must be performed within 12 hours after IRMs are on Range 2 or below. The allowance to enter the Applicability with the 31 day Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. Twelve hours is reasonable based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances.

SR 3.3.1.2.7

Performance of a CHANNEL CALIBRATION at a Frequency of 92 days verifies the performance of the SRM monitors 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 neutron detectors are excluded from the CHANNEL CALIBRATION (Note 1) because they cannot readily be adjusted. The detectors are fission chambers that are designed to have a relatively constant sensitivity over the range and with an accuracy specified for a fixed useful life.

Note 2 to the Surveillance allows the Surveillance to be delayed until entry into the specified condition of the Applicability. The SR must be performed in MODE 2 within 12 hours of entering MODE 2 with IRMs on Range 2 or below. The allowance to enter the Applicability with the 92 day

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2.7 (continued)

Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. Twelve hours is reasonable based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances.

REFERENCES

None.

TECHNICAL CHANGES - MORE RESTRICTIVE

M2 An additional Function has been added to CTS Table 3.2-3. ITS 3.3.2.1, Control Rod Block Instrumentation, will include the Control Rod Block Function of the Reactor Mode Switch as a required function (Function 3 on proposed Table 3.3.2.1-1). The new requirement is that 2 channels of the Rod Block function of Reactor Mode Switch – Shutdown Position must be Operable whenever the Mode Switch is in the Shutdown position. This addition to the Specification for the Control Rod Block Instrumentation will include proposed SR 3.3.2.1.7 (CHANNEL FUNCTIONAL TEST every 24 months) and proposed LCO 3.3.2.1, Condition E (Required Actions and Completion Times if this function is inoperable). ITS SR 3.3.2.1.7 will not be required to be performed until 1 hour after the Reactor Mode Switch is placed in Shutdown. This rod block ensures that control rods are not withdrawn in MODES 3 and 4, since control rods are assumed to be inserted. This change is consistent with NUREG-1433, Revision 1.

M3 The out of service time in CTS Table 3.2-3 Note 2 Action B.a) has been reduced from 7 days to 24 hours (ITS 3.3.2.1 Required Action A.1) when one RBM channel is inoperable. The 24 hour Completion Time is acceptable, based on a low probability of an event occurring coincident with a failure in the remaining channel. This change is more restrictive since less time is permitted but consistent with NUREG-1433, Revision 1.

M4 SR 3.3.2.1.4 has been added to CTS Table 4.2.3 to verify that the RBM is not bypassed at Thermal Power ≥ 30% RTP and when a peripheral control rod is not selected every 92 days. This change is more restrictive since a periodic surveillance has been included. This will ensure the RBM is Operable when required to limit the consequences of a single control rod withdrawal error event during power operation.

M5 A new CHANNEL FUNCTIONAL TEST (ITS SR 3.3.2.1.3) surveillance is proposed to be added similar to CTS 4.3.B.3.a.4 in MODE 1 when Thermal Power is ≤ 10% to ensure the RWM is Operable with the reactor mode switch in RUN. The test is required every 92 days and is consistent with NEDC-30851-P-A, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

M6 A new SR is proposed to be added to the surveillances of CTS 4.3.B.3. SR 3.3.2.1.6 will verify every 24 months that the Rod Worth Minimizer (RWM) is not bypassed when Thermal Power is ≤ 10%. The RWM may be bypassed when power is above 10%. However, the existing specifications (CTS 4.3.B.3) do not have an explicit requirement to verify the setpoint of the RWM bypass feature. This change represents an additional restriction on plant operations necessary to ensure the RWM Function is Operable when required.

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
[M2] SR 3.3.2.1.~~6~~ 7	DB4 ------NOTE------ Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. ---------- Perform CHANNEL FUNCTIONAL TEST.	~~18~~ 24 months DB7
move to previous page [L5] [Table 4.2-3 function 6 and 7] SR 3.3.2.1.~~7~~ 5	and the recirculation loop flow signals ------NOTE------ Neutron detectors are excluded. CLB1 ---------- Perform CHANNEL CALIBRATION.	92 days ~~18 months~~ DB4 J
[4.3.B.3.a.1] [4.3.B.3.b.1] [L4] SR 3.3.2.1.8	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM

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RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 ITS SR 3.3.2.1.5, the 92 day RBM Channel Calibration Surveillance, is modified by the addition of a Note that excludes the recirculation loop flow signal portion of the channel. CTS Table 4.1-2, "Flow Bias Signal," requires an "internal power and flow test with standard pressure source" calibration on a refueling interval. This is covered by ITS SR 3.3.1.1.12. This flow biased signal provides input to both the APRM Neutron Flux – High (Flow Biased) RPS scram Function and to the RBM – Upscale control rod block Function. CTS 3/4.2.C does not have a specific flow bias signal line item, thus the calibration required by CTS Table 4.1-2 covers the RBM requirements, as well as the RPS requirements, of the flow bias signal. Therefore, the RBM Channel Calibration requirement in SR 3.3.2.1.5 is modified to exclude the flow bias signal. The ITS Bases clearly identifies that SR 3.3.1.1.12 covers the flow bias signal.

J

CLB2 The Allowable Value of the RBM upscale is located in the COLR. This was accepted in JAFNPP Technical Specification Amendment No.162. This allowance is consistent with the guidance in Generic Letter 88-16 for the removal of cycle-specific parameter limits from the Technical Specifications to the COLR.

CLB3 The CTS allows only one startup with the RWM inoperable (i.e., inoperable prior to withdrawal of the first 12 control rods) per calendar year. The words in ISTS Required Action C.2.1.2, "performed in the last calendar year" could allow multiple startups with the RWM inoperable in the current calendar year, since the check only looks at the last (i.e., previous) calendar year. Therefore, consistent with the current licensing basis, the word "last" has been changed to "current."

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 None

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The RWM is required to be Operable at ≤ 10% RTP as specified in CTS 4.3.B.3.a.4. This requirement is consistent with the design bases analysis assumptions. Therefore, the bracketed value of 10% has been retained in the ITS throughout the Specification.

DB2 The brackets have been removed and the Surveillance Frequency of 92 days is retained in ITS SR 3.3.2.1.2 and SR 3.3.2.1.3. This Frequency is justified in DOC L3.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB3 ITS SR 3.3.2.1.4 has been added in accordance with M4. The bracketed Frequency of 18 months has been changed to 92 days and the bracketed Surveillance Note (Neutron detectors are excluded) retained. The surveillance has been re-written to conform to the JAFNPP plant design. The Surveillance ensures the RBM is Operable when required.

DB4 ISTS SR 3.3.2.1.7, (Channel Calibration of RBM Upscale & Downscale channels) is currently performed every 92 days therefore the surveillance has been placed in its appropriate location and renumbered as SR 3.3.2.1.5. Subsequent surveillances have been renumbered, where applicable. This Surveillance Frequency is consistent with methodology in determining the associated Allowable Values for these Functions. Since the Calibration is performed every 92 days there is no need for a CHANNEL FUNCTIONAL TEST, therefore SR 3.3.2.1.1 has been removed from these Functions in the Table.

DB5 SR 3.3.2.1.1, a CHANNEL FUNCTIONAL TEST, has been added in accordance with M1 for the RBM Inop function. The bracketed Frequency of 92 days is retained since it is consistent with NEDC-30851-P-A.

DB6 The bracketed Surveillance Frequency of ITS SR 3.3.2.1.6 is changed from 18 months to 24 months as justified in the associated Bases for this surveillance. The trip setpoint methodology assumes a Frequency of 24 months between calibrations.

DB7 The bracketed Surveillance Frequency of ITS SR 3.3.2.1.7 has been changed from 18 to 24 months since the test should be performed during a plant outage to minimize any unplanned transients as described in the Bases for this SR.

DB8 The brackets have been removed and the proper number of channels included for each Function in Table 3.3.2.1-1. The values are consistent with the current requirements in CTS Table 3.2.3 for Functions 1.a, 1.c, and CTS 3.3.B.3 for the Rod Worth Minimizer. The requirements for Function 1.b (RBM-Inop) and Function 3 (Reactor Mode Switch-Shutdown) have been added in accordance with M1 and M2. The specified number of channels are consistent with the plant design.

DB9 Table 3.3.2.1-1 Functions 1.b, 1.c and 1.f are not applicable to JAFNPP. Therefore these Functions have been removed from the Table. Subsequent Functions have been renumbered, where applicable.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.3.2.1 - CONTROL ROD BLOCK INSTRUMENTATION

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB10 The Table 3.3.2.1-1 Applicability for the RBM Functions have been revised to be consistent with the JAFNPP plant design. The RBM setpoint includes three different curves which vary as a Function of recirculation flow. The Allowable Values for these Functions are included in the COLR since they vary depending on the cycle. All three curves must be Functioning properly whenever Thermal Power is ≥ 30% RTP and when no peripheral control rod is selected. Therefore Table Footnotes b, c, d, and e have been deleted and (a) revised to meet the specific JAFNPP Applicability. Subsequent Applicability Footnotes have been revised, as required.

DB11 The brackets have been removed and the "Allowable Value" included consistent with the requirements in CTS Table 3.2-3.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The change presented in the Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Editorial Changes Affecting NUREG-1433 designated as NRC-ED-14 has been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued) 10 CFR 50.36(c)(2)(ii) (Ref. 4) X1

The RBM Function satisfies Criterion 3 of ~~the NRC Policy Statement~~.

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value ~~for the associated power range~~ specified in the COLR CLB3, to ensure that no single ~~instrument~~ failure can preclude a rod block from this Function. The actual setpoints are calibrated consistent with applicable setpoint methodology. DB1 PA2

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. ~~The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.~~

Insert ASA DB6

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ ~~29%~~ 30 RTP and a peripheral control rod is not selected. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. ~~8~~ 1) DA2. ~~When operating < 90% RTP, analyses (Ref. 3) have shown that with an initial MCPR ≥ 1.70, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR ≥ 1.40, no RWE event will result in exceeding the MCPR~~ or if a peripheral control rod is selected, DB1 1J

(continued)

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BASES

ACTIONS

C.1, C.2.1.1, C.2.1.2, and C.2.2 (continued)

The RWM may be bypassed under these conditions to allow continued operations. In addition, Required Actions of LCO 3.1.3 and LCO 3.1.6 may require bypassing the RWM, during which time the RWM must be considered inoperable with Condition C entered and its Required Actions taken.

D.1

With the RWM inoperable during a reactor shutdown, the operator is still capable of enforcing the prescribed control rod sequence. Required Action D.1 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff. (i.e., reactor engineer) CLB1
The RWM may be bypassed under these conditions to allow the reactor shutdown to continue.

E.1 and E.2

With one Reactor Mode Switch—Shutdown Position control rod withdrawal block channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod withdrawal block function. However, since the Required Actions are consistent with the normal action of an OPERABLE Reactor Mode Switch—Shutdown Position Function (i.e., maintaining all control rods inserted), there is no distinction between having one or two channels inoperable.

In both cases (one or both channels inoperable), suspending all control rod withdrawal and initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies will ensure that the core is subcritical with adequate SDM ensured by (LCO 3.1.1). Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are therefore not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

, "SHUTDOWN MARGIN (SDM)" 5 PA1

(continued)

Revision J

TA1

INSERT SR 3.3.2.1.1

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with the applicable extensions. J

DB8

Testing of the Reactor Manual Control Multiplexing System input shall include inputs of "no control rod selected," "peripheral control rod selected," and other control rods selected with two, three, or four LPRM assemblies around it. In addition, testing shall include a verification that an inoperable trip occurs when a module is not plugged in, or the function switch is moved to any position other than "Operate".

TA1

INSERT SR 3.3.2.1.2

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with the applicable extensions. J

TA1 INSERT SR 3.3.2.1.7

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with the applicable extensions. 1 J

BASES

SURVEILLANCE REQUIREMENTS

DB4

move to page B 3.3-52

SR 3.3.2.1.~~6~~ 5 (continued)

adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.~~8~~ 9. PA5

Also as noted, the recirculation loop flow signals are excluded since this portion of the channel is calibrated by SR 3.3.1.1.12. PA4

The Frequency is based upon the assumption of a~~n~~ ~~18 month~~ 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. DB4

J

SR 3.3.2.1.8

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

DB2

PA6 DB7

REFERENCES

DB2 X1 renumbering

1. U~~F~~SAR, Section ~~7.6.2.2.5~~ 5.8.2.

2. U~~F~~SAR, Section ~~7.6.8.2.6~~ 16.5.3.

3. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuels, Supplement for United States, Section S.2.2.1.5, (Revision specified in the COLR).

4. 10 CFR 50.36(c)(2)(ii)

X1 J

~~3. NEDC-30474-P, "Average Power Range Monitor, Rod Block Monitor, and Technical Specification Improvements (ARTS) Program for Edwin I. Hatch Nuclear Plants," December 1983.~~

DB2

5. UFSAR, Section 14.6.1.2

~~4. NEDE-24011-P-A-9-US, "General Electrical Standard Application for Reload Fuel," Supplement for United States, Section S 2.2.3.1, September 1988.~~

DB2

~~5.~~ 7. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners' Group, July 1986.

Insert from next page DB2

Letter from T. A. Pickens (BWROG) to G. C. Lainas (NRC), Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A, BWROG-8644, August 15, 1986.

(continued)

J

Revision J

BASES

DB2

X1
renumbering

move to previous page

DB2

J

REFERENCES (continued)

~~6. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.~~

6 ~~7~~. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987. PA7

9 ~~8~~. NEDC-30851-P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988. A

8 ~~9~~. GENE-770-06-1, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," PA7 ~~February 1991~~ December 1992.

Revision J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.3.2.1 - CONTROL ROD BLOCK INSTRUMENTATION

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The reactor engineers are the only other persons qualified at JAFNPP to verify movement of control rods therefore the phrase "(i.e., reactor engineer)" has been added to describe the "other qualified member of the technical staff" in ITS 3.3.2.1 Required Actions C.2.2 and D.1 Bases.

CLB2 This requirement to prepare a report is consistent with the current requirements in CTS 3.3.B.3.d.

CLB3 The Allowable Value of the RBM upscale is located in the COLR. This was accepted in JAFNPP Technical Specification Amendment No.162. This allowance is consistent with the guidance in Generic Letter 88-16 for the removal of cycle-specific parameter limits from the Technical Specifications to the COLR.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 The Bases have been revised for clarity or accuracy.

PA2 The Bases have been revised to be consistent with the other places in the Bases.

PA3 The Reviewer's Note has been deleted.

PA4 The Bases has been revised to be consistent with the Specification.

PA5 The appropriate Surveillance has been included as a result of changes made to the Surveillances of ITS 3.3.1.1.

PA6 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.

PA7 The quotations used in the Bases References have been removed. The Writer's Guide does not require the use of quotations.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific design.

DB2 The appropriate references have been included. Subsequent References have been renumbered, as applicable.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.3.2.1 - CONTROL ROD BLOCK INSTRUMENTATION

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB3 The Frequency of ITS SR 3.3.2.1.4 has been changed from 18 months to 92 days consistent with the setpoint methodology for the associated channels.

DB4 ISTS SR 3.3.2.1.7 has been renumbered as SR 3.3.2.1.5 since the setpoint methodology for the Rod Block Monitor - Upscale trip is based on a Surveillance Frequency of 92 days instead of 24 months. Subsequent SRs have been renumbered, as applicable.

DB5 The Surveillance Frequency of ITS SR 3.3.2.1.7 has been changed from 18 to 24 months since the test should be performed during a plant outage to minimize any unplanned transients as described in the Bases.

DB6 The Bases description of the setpoint methodology has been revised to be consistent with the JAFNPP plant specific methodology.

DB7 The brackets have been removed and the appropriate Plant Specific References included.

DB8 The Bases of ISTS SR 3.3.2.1.1 has been changed to account for changes made to the Specification. The only Function tested under this Surveillance is the RBM inop circuitry. The Bases describes the actual testing required by the Surveillance.

DB9 Existing Reference 5 (ITS Reference 7) is actually an attachment to another document. The actual reference has been revised to reflect this other document in order to facilitate location of the references in the future. J

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 205, Revision 3 have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 NUREG-1433, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995. References have been renumbered, as applicable.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.2	-----------------NOTE----------------- Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	-----------------NOTE----------------- Not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	-----------------NOTE----------------- Neutron detectors are excluded. -- Verify the RBM is not bypassed: a. When THERMAL POWER is ≥ 30% RTP; and b. When a peripheral control rod is not selected.	92 days
SR 3.3.2.1.5	-----------------NOTE----------------- Neutron detectors and the recirculation loop flow signals are excluded. -- Perform CHANNEL CALIBRATION.	92 days

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value specified in the COLR, to ensure that no single failure can preclude a rod block from this Function. The actual setpoints are calibrated consistent with applicable setpoint methodology.

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The trip setpoints are derived from the analytical limits and account for all worst case instrumentation uncertainties as appropriate (e.g., drift, process effects, calibration uncertainties, and severe environmental errors (for channels that must function in harsh environments as defined by 10 CFR 50.49)). The trip setpoints derived in this manner provide adequate protection because all expected uncertainties are accounted for. The Allowable Values are then derived from the trip setpoints by accounting for normal effects that would be seen during periodic surveillance or calibration. These effects are instrumentation uncertainties observed during normal operation (e.g., drift and calibration uncertainties).

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 30% RTP and a peripheral control rod is not selected. Below this power level, or if a peripheral control rod is selected, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 1). | 5

(continued)

BASES

ACTIONS

C.1, C.2.1.1, C.2.1.2, and C.2.2 (continued)

The RWM may be bypassed under these conditions to allow continued operations. In addition, Required Actions of LCO 3.1.3 and LCO 3.1.6 may require bypassing the RWM, during which time the RWM must be considered inoperable with Condition C entered and its Required Actions taken.

D.1

With the RWM inoperable during a reactor shutdown, the operator is still capable of enforcing the prescribed control rod sequence. Required Action D.1 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff (i.e., reactor engineer). The RWM may be bypassed under these conditions to allow the reactor shutdown to continue.

E.1 and E.2

With one Reactor Mode Switch – Shutdown Position control rod withdrawal block channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod withdrawal block function. However, since the Required Actions are consistent with the normal action of an OPERABLE Reactor Mode Switch – Shutdown Position Function (i.e., maintaining all control rods inserted), there is no distinction between having one or two channels inoperable.

In both cases (one or both channels inoperable), suspending all control rod withdrawal and initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies will ensure that the core is subcritical with adequate SDM, (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"). Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are therefore not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted. J

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted (Note 1) at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are modified by Note 2 to indicate that when an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 8) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control Multiplexing System input. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with the applicable extensions. Testing of the Reactor Manual Control Multiplexing System input shall include inputs of "no control rod selected," "peripheral control rod selected," and other control rods selected with two, three, or four LPRM assemblies around it. In addition, testing shall include a verification that an inoperable trip occurs when a module is not plugged in, or the function switch is moved to any position other than "Operate". The Frequency of 92 days is based on reliability analyses (Ref. 9).

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.5 (continued)

range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7. Also as noted, the recirculation loop flow signals are excluded since this portion of the channel is calibrated by SR 3.3.1.1.12.

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

SR 3.3.2.1.6

The RWM is automatically bypassed when power is above a specified value. The power level is determined from steam flow signals compensated for steam pressure. The automatic bypass setpoint must be verified periodically to be ≤ 10% RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on the trip setpoint methodology utilized for the low power setpoint channel.

SR 3.3.2.1.7

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch-Shutdown Position Function to ensure that the entire channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with the applicable extensions. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch-Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.7 (continued)

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into MODES 3 and 4 if the 24 month Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.2.1.8

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

REFERENCES

1. UFSAR, Section 7.5.8.2.
2. UFSAR, Section 7.16.5.3.
3. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, Supplement for United States, Section S.2.2.1.5, (Revision specified in the COLR). J
4. 10 CFR 50.36(c)(2)(ii).
5. UFSAR, Section 14.6.1.2.
6. NRC SER, Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17, December 27, 1987.
7. Letter from T.A. Pickens (BWROG) to G.C. Lainas (NRC), Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A, BWROG-8644, August 15, 1986. J

(continued)

BASES

REFERENCES (continued)

8. GENE-770-06-1-A, Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications, December 1992.

9. NEDC-30851P-A, Supplement 1, Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation, October 1988.

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

[T.3.2-6(1)] [T.3.2-6 Note 1]
[T.4.2-6(1)]
[3.2.F]

LCO 3.3.2.2 [Three] channels of feedwater and main turbine high water level trip instrumentation shall be OPERABLE. DB1

[TABLE 3.2-6]

APPLICABILITY: THERMAL POWER ≥ [25]% RTP. DB2

ACTIONS

[A3]

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
[Table 3.2-6 Note 1a] A. One feedwater and main turbine high water level trip channel inoperable.	A.1 Place channel in trip.	7 days
[Table 3.2-6 Note 1.b] B. Two or more feedwater and main turbine high water level trip channels inoperable.	B.1 Restore feedwater and main turbine high water level trip capability.	2 hours
[Table 3.2-6 Notes 1.a & 1.b] C. Required Action and associated Completion Time not met.	C.~~1~~ 2 TA1 Reduce THERMAL POWER to < [25]% RTP.	4 hours DB2

PA1 C.1 TA1

----- NOTE -----
Only applicable if inoperable channel is the result of inoperable feedwater pump [Valve] or main turbine stop valve.
turbine DB6

PA1 ~~C.~~1 Remove affected ~~feedwater pump(s) and main turbine~~ Valve(s) from service. 4 hours [J]
Stop DB6 TA1

OR

BASES (continued)

APPLICABILITY

The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," sufficient margin to these limits exists below 25% RTP; therefore, these requirements are only necessary when operating at or above this power level.

PA 2

J

ACTIONS

A Note has been provided to modify the ACTIONS related to feedwater and main turbine high water level trip instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable feedwater and main turbine high water level trip instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable feedwater and main turbine high water level trip instrumentation channel.

A.1

With one channel inoperable, the remaining two OPERABLE channels can provide the required trip signal. However, overall instrumentation reliability is reduced because a single failure in one of the remaining channels concurrent with feedwater controller failure, maximum demand event, may result in the instrumentation not being able to perform its intended function. Therefore, continued operation is only allowed for a limited time with one channel inoperable. If the inoperable channel cannot be restored to OPERABLE status within the Completion Time, the channel must be placed in the tripped condition per Required Action A.1. Placing the

(continued)

Revision J

TA2 INSERT SR 3.3.2.2-1

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. 1

CLB1 INSERT SR 3.3.2.2.2-2

As noted, the CHANNEL FUNCTIONAL TEST is only required to be performed when in MODE 4 for > 24 hours. In MODE 4, the plant is in a condition where a loss of a feedwater pump turbine or a main turbine trip will not jeopardize steady state power operation. The design of the trip systems do not permit functional testing of this trip function without lifting electrical leads. Consequently, testing the trip systems on-line poses an unacceptable risk of an inadvertent trip of the feedwater pump turbines and main turbine, resulting in a plant transient. The 24 hours is intended to indicate an outage of sufficient duration to allow for scheduling a proper performance of the Surveillance.

The 92 day Frequency and the Note to this Surveillance are based on Reference 5.

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Three channels of feedwater and main turbine high water level trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One feedwater and main turbine high water level trip channel inoperable.	A.1	Place channel in trip.	7 days
B. Two or more feedwater and main turbine high water level trip channels inoperable.	B.1	Restore feedwater and main turbine high water level trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1	--------NOTE-------- Only applicable if inoperable channel is the result of inoperable feedwater pump turbine or main turbine stop valve. -------------------- Remove affected stop valve(s) from service.	4 hours
	<u>OR</u>		
	C.2	Reduce THERMAL POWER to < 25% RTP.	4 hours

1 (J)

BASES

LCO (continued)

uncertainties, and severe environmental errors (for channels that must function in harsh environments as defined by 10 CFR 50.49)). The trip setpoints derived in this manner provide adequate protection because all expected uncertainties are accounted for. The Allowable Values are then derived from the trip setpoints by accounting for normal effects that would be seen during periodic surveillance or calibration. These affects are instrumentation uncertainties during normal operation (e.g., drift and calibration uncertainties).

APPLICABILITY

The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," sufficient margin to these limits exists below 25% RTP; therefore, these requirements are only necessary when operating at or above this power level. J

ACTIONS

A Note has been provided to modify the ACTIONS related to feedwater and main turbine high water level trip instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable feedwater and main turbine high water level trip instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable feedwater and main turbine high water level trip instrumentation channel.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the feedwater pump turbines and main turbine will trip when necessary.

SR 3.3.2.2.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between instrument channels could be an indication of excessive instrument drift in one of the channels, or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Channel agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limits.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel status during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.2.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.2.2 (continued)

As noted, the CHANNEL FUNCTIONAL TEST is only required to be performed when in MODE 4 for > 24 hours. In MODE 4, the plant is in a condition where a loss of a feedwater pump turbine or a main turbine trip will not jeopardize steady state power operation. The design of the trip systems do not permit functional testing of this trip function without lifting electrical leads. Consequently, testing the trip systems on-line poses an unacceptable risk of an inadvertent trip of the feedwater pump turbines and main turbine, resulting in a plant transient. The 24 hours is intended to indicate an outage of sufficient duration to allow for scheduling a proper performance of the Surveillance.

The 92 day Frequency and the Note to this Surveillance are based on Reference 5.

SR 3.3.2.2.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency 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 3.3.2.2.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the feedwater and main turbine valves is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a valve is incapable of operating, the associated instrumentation would also be inoperable. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 14.5.9.

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

3. Drawing 11825-5.01-15D, Rev. D, Reactor Assembly Nuclear Boiler, (GE Drawing 919D690BD).

4. GENE-770-06-1-A, Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications, December 1992.

5. NRC letter dated June 19, 1995, Amendment 225 for James A. FitzPatrick Nuclear Power Plant.

OBC INSERT LCO-5

Two physically separated and redundant radiation detectors with a range of 1 R/hr to 1E8 R/hr are located inside the drywell. The detectors provide a signal to separate process radiation monitors located in the control room. These radiation detectors and associated monitors provide the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with these portions of the instrument channel. J

CLB1 INSERT LCO-6

6. Drywell Temperature

Drywell temperature is a Category 1 variable provided to detect a breach in the RCPB and to verify ECCS functions that operate to maintain RCS integrity. Two drywell temperature channels monitor the range from 40°F to 440°F. Each drywell temperature channel consists of a separate temperature sensor, with an associated indicator and recorder in the control room. These temperature sensors and associated recorders provide the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with these portions of the instrument channel.

X5 INSERT Footnote (a)

Therefore, this Function is not required for isolation valves whose associated penetration flow path is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured (as noted in footnote (a) to Table 3.3.3.1-1).

DB1 INSERT LCO-7

The PCIV position PAM instrumentation consists of position switches mounted on the valves for the positions to be indicated, associated wiring and control room indicating lamps for active PCIVs (check valves and manual valves are not required to have position indication). These position switches and associated indicators in the control room provide the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with these portions of the instrument channel.

DB1 INSERT LCO-8

The primary containment hydrogen and oxygen concentration channels consists of two redundant analyzers. Each analyzer contains a hydrogen and an oxygen detector. Each analyzer can be aligned to sample air from one of four sample points (3 points in the drywell and 1 point in the suppression chamber). Sample air passes through the hydrogen analyzer and the oxygen analyzer and is returned to the suppression chamber air space. During normal operation, the Division I analyzer samples the suppression chamber and the Division II analyzer samples the drywell. The analyzers are capable of determining oxygen and hydrogen concentrations in the range of 0% to 30%, which meets the requirements of Reference 1. The hydrogen and oxygen concentration from each analyzer may be displayed on its associated recorder in the relay room. Therefore, the PAM Specification deals specifically with these portions of the instrument channel. A Note allows the primary containment hydrogen and oxygen concentration channels to be inoperable for up to 3 hours per 24 hour period during Post Accident Sampling System (PASS) operation. PASS operation includes realignment from or to the mode. Operation of the PASS may require isolation of the primary containment hydrogen and oxygen concentration channels. This allowance will ensure that the PASS can perform its post accident monitoring function (Ref. 4) while minimizing the time the primary containment hydrogen and oxygen concentration channels are isolated.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1.1 (continued)

gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. ~~The high radiation instrumentation should be compared to similar plant instruments located throughout the plant.~~ X4

For the PCIV Position Function, the CHANNEL CHECK consists of verifying the remote indication conforms to expected valve position.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. Channel PA2

X4 For the PCIV Position Function, the CHANNEL CALIBRATION consists of verifying the remote indication conforms to actual valve position.

The Frequency of 31 days is based upon plant operating experience, with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 31 day interval is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of those displays associated with the ~~required~~ channels ~~of this~~ by the LCO. PA1

SR 3.3.3.1.2 and SR 3.3.3.1.3 CLB2

CLB2 These SRs require A CHANNEL CALIBRATION ~~is~~ to be performed ~~every 18 months, or approximately at every refueling~~. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies the channel responds to measured parameter with the necessary range and accuracy. CLB2

The 92 day Frequency for CHANNEL CALIBRATION of the Primary Containment Hydrogen and Oxygen Concentration channels is based on vendor recommendations. CLB2

for CHANNEL CALIBRATION of all other PAM instrumentation of Table 3.3.3.1-1 J

The 24 month DBL Frequency is based on operating experience and consistency with the ~~typical industry~~ refueling cycles. PA1

REFERENCES

1. Regulatory Guide 1.97, Revision 3, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," ~~[date]~~ May 1983. PA5 CLB3 J

2. ~~[Plant specific documents (e.g., NRC Regulatory Guide 1.97, SER letter).]~~ INSERT REF CLB3

Revision J

CLB3

INSERT REF

2. NRC letter, H. I. Abelson to J. C. Brons dated March 14, 1988, regarding conformance to Regulatory Guide 1.97, Rev. 2. Includes NRR Safety Evaluation Report for Regulatory Guide 1.97 and James A. FitzPatrick Nuclear Power Plant.

3. 10 CFR 50.36(c)(2)(ii).

4. UFSAR Section 9.14.4.

5. DRF-T23-688-1, Error in FitzPatrick Temperature Measurement Based on Monticello In-plant S/RV Test Data.

BASES

LCO (continued)

The following list is a discussion of the specified instrument Functions listed in Table 3.3.3.1-1 in the accompanying LCO.

1. Reactor Vessel Pressure

Reactor vessel pressure is a Category 1 variable provided to support monitoring of Reactor Coolant System (RCS) integrity and to verify operation of the Emergency Core Cooling Systems (ECCS). Two independent pressure transmitters with a range of 0 psig to 1500 psig monitor pressure and associated independent wide range recorders are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with these portions of the instrument channel.

2. Reactor Vessel Water Level

Reactor vessel water level is a Category 1 variable provided to support monitoring of core cooling and to verify operation of the ECCS. The reactor vessel water level channels provide the PAM Reactor Vessel Water Level Function. The reactor vessel water level channels cover a range from -150 inches (just below the bottom of the active fuel) to +224.5 inches, as referenced (zero) from the top of active fuel (TAF). Reactor vessel water level is measured in overlapping stages by separate independent differential pressure transmitters. Two reactor vessel water level (fuel zone) channels monitor the range from -150 inches to +200 inches (TAF). One fuel zone channel consists of a transmitter and indicator and the other channel consists of a transmitter and recorder. Two reactor vessel water level (wide range) channels monitor the range from +14.5 inches to +224.5 inches (TAF). The upper limit corresponds to a level of 63.5 inches below the centerline of the main steam lines. Likewise, one wide range channel consists of a transmitter and indicator and the other channel consists of a transmitter and recorder. These transmitters and associated indicators and recorders provide the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with these portions of the instrument channel.

(continued)

BASES

LCO (continued)

5. Containment High Range Radiation

Containment high range radiation channels are provided to monitor the potential of significant releases of radioactive material and to provide release assessment for use by operators in determining the need to invoke site emergency plans. Two physically separated and redundant radiation detectors with a range of 1 R/hr to 1E8 R/hr are located inside the drywell. The detectors provide a signal to separate process radiation monitors located in the control room. These radiation detectors and associated monitors provide the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with these portions of the instrument channel.

6. Drywell Temperature

Drywell temperature is a Category 1 variable provided to detect a breach in the RCPB and to verify ECCS functions that operate to maintain RCS integrity. Two drywell temperature channels monitor the range from 40°F to 440°F. Each drywell temperature channel consists of a separate temperature sensor, with an associated recorder in the control room. These temperature sensors and associated recorders provide the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with these portions of the instrument channel.

7. Primary Containment Isolation Valve (PCIV) Position

PCIV position is a Category 1 variable provided for verification of containment integrity. In the case of PCIV position, the important information is the isolation status of the containment penetration. Therefore, this Function is not required for isolation valves whose associated penetration flow path is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured (as noted in footnote (a) to Table 3.3.3.1-1). The LCO requires one channel of valve position indication in the control room to be OPERABLE for each active PCIV in a containment penetration flow path, i.e., two total channels of PCIV position indication for a penetration flow path with

(continued)

BASES

LCO

7. Primary Containment Isolation Valve (PCIV) Position (continued)

two active valves. For containment penetrations with only one active PCIV having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration via indicated status of the active valve, as applicable, and prior knowledge of passive valve or system boundary status. If a penetration flow path is isolated, position indication for the PCIV(s) in the associated penetration flow path is not needed to determine status. Therefore, the position indication for valves in an isolated penetration flow path is not required to be OPERABLE. Each penetration is treated separately and each penetration flow path is considered a separate Function. Therefore, separate Condition entry is allowed for each inoperable penetration flow path.

The PCIV position PAM instrumentation consists of position switches mounted on the valves for the positions to be indicated, associated wiring and control room indicating lamps for active PCIVs (check valves and manual valves are not required to have position indication). These position switches and associated indicators in the control room provide the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with these portions of the instrument channel.

8. Primary Containment Hydrogen and Oxygen Concentration

Primary containment hydrogen and oxygen concentration is a Category 1 variable provided to detect high hydrogen or oxygen concentration conditions that represent a potential for containment breach. This variable is also important in verifying the adequacy of mitigating actions. The primary containment hydrogen and oxygen concentration channels consists of two redundant analyzers. Each analyzer contains a hydrogen and an oxygen detector. Each analyzer can be aligned to sample air from one of four sample points (3 points in the drywell and 1 point in the suppression chamber). Sample air passes through the hydrogen analyzer and the oxygen analyzer and is returned to the suppression chamber air space. During normal operation, the Division I

(continued)

BASES

LCO

8. Primary Containment Hydrogen and Oxygen Concentration (continued)

analyzer samples the suppression chamber and the Division II analyzer samples the drywell. The analyzers are capable of determining oxygen and hydrogen concentrations in the range of 0% to 30%, which meets the requirements of Reference 1. The hydrogen and oxygen concentration from each analyzer may be displayed on its associated recorder in the relay room. Therefore, the PAM Specification deals specifically with these portions of the instrument channel. A Note allows the primary containment hydrogen and oxygen concentration channels to be inoperable for up to 3 hours per 24 hour period during Post Accident Sampling System (PASS) operation. PASS operation includes realignment from or to the mode. Operation of the PASS may require isolation of the primary containment hydrogen and oxygen concentration channels. This allowance will ensure that the PASS can perform its post accident monitoring function (Ref. 4) while minimizing the time the primary containment hydrogen and oxygen concentration channels are isolated.

9. Suppression Chamber Pressure

Suppression chamber pressure is a Category 1 variable provided to verify RCS and containment integrity and to verify the effectiveness of ECCS actions taken to prevent containment breach. Two suppression chamber channels monitor a range from -15 psig to +85 psig. Each channel consists of an independent transmitter and associated recorder in the control room. These transmitters and recorders provide the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with these portions of the instrument channel.

10. Suppression Pool Water Temperature

Suppression pool water temperature is a Category 1 variable provided to detect a condition that could potentially lead to containment breach and to verify the effectiveness of ECCS actions taken to prevent containment breach. The suppression pool water temperature instrumentation allows operators to detect trends in suppression pool water temperature. The suppression pool water temperature is monitored by two redundant channels. Each channel consists of sixteen resistance temperature detectors (RTDs) that

(continued)

BASES

LCO

10. Suppression Pool Water Temperature (continued)

monitor temperature over a range of 30°F to 230°F. The RTDs are mounted in thermowells spaced at equal intervals around the periphery of the suppression pool. The sixteen RTD signals are averaged and the resulting bulk temperature signal is sent to redundant indicating recorders in the control room. A minimum of fifteen out of sixteen RTDs are required for channel operability. An evaluation (Ref. 5) demonstrates that the maximum error in suppression pool bulk temperature measurement including channel uncertainty is < 4°F with active pool circulation. Thus a 4°F bias has been employed for conservatism. By specifying 15 RTDs the single failure criteria is accounted for. This evaluation conservatively assumed the failure of RTDs at locations that minimized indicated bulk suppression pool temperature and consequently maximized indicated error. These RTDs and recorders provide the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with these portions of the instrument channels.

11. Drywell Water Level

Drywell Water Level is a Category 1 variable provided to detect whether plant safety functions are being accomplished. Two drywell water level channels monitor the range from 22 feet to 106 feet. Each drywell water level channel consists of level transmitters, with an associated indicator and recorder in the control room. These level transmitters and associated indicators and recorders provide the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with these portions of the instrument channel.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1 and 2. These variables are related to the diagnosis and preplanned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1 and 2. In MODES 3, 4, and 5, plant conditions are such that the likelihood of an event that would require PAM instrumentation is extremely low; therefore, PAM instrumentation is not required to be OPERABLE in these MODES.

(continued)

BASES

ACTIONS

Note 1 has been added to the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to diagnose an accident using alternative instruments and methods, and the low probability of an event requiring these instruments.

Note 2 has been provided to modify the ACTIONS related to PAM instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable PAM instrumentation channels provide appropriate compensatory measures for separate Functions. As such, a Note has been provided that allows separate Condition entry for each inoperable PAM Function.

A.1

When one or more Functions have one required channel that is inoperable, the required inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channels (or, in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is initiated by these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

If a channel has not been restored to OPERABLE status in 30 days, this Required Action specifies initiation of action in accordance with Specification 5.6.6, which requires a written report to be submitted to the NRC. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1.1 (continued)

Channel agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency of 31 days is based upon plant operating experience, with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 31 day interval is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of those displays associated with the channels required by the LCO.

SR 3.3.3.1.2 and SR 3.3.3.1.3

These SRs require a CHANNEL CALIBRATION to be performed. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies the channel responds to measured parameter with the necessary range and accuracy. For the PCIV Position Function, the CHANNEL CALIBRATION consists of verifying the remote indication conforms to actual valve position.

The 92 day Frequency for CHANNEL CALIBRATION of the Primary Containment Hydrogen and Oxygen Concentration channels is based on vendor recommendations. The 24 month Frequency for CHANNEL CALIBRATION of all other PAM instrumentation of Table 3.3.3.1-1 is based on operating experience and consistency with the refueling cycle.

REFERENCES

1. Regulatory Guide 1.97, Revision 3, Instrumentation For Light-Water-Cooled Nuclear Power Plants To Assess Plant And Environs Conditions During And Following An Accident, May 1983.

2. NRC letter, H. I. Abelson to J. C. Brons dated March 14, 1988, regarding conformance to Regulatory Guide 1.97, Rev. 2. Includes NRR Safety Evaluation Report for Regulatory Guide 1.97 and James A. FitzPatrick Nuclear Power Plant.

(continued)

BASES

REFERENCES (continued)

3. 10 CFR 50.36(c)(2)(ii).
4. UFSAR, Section 9.14.4.
5. DRF-T23-688-1, Error in FitzPatrick Temperature Measurement Based on Monticello In-plant S/RV Test Data. (J

1 J

BASES

DB2 and 2

DB2

APPLICABLE SAFETY ANALYSES (continued)

The criteria governing the design and the specific system requirements of the Remote Shutdown System are located in ~~10 CFR 50, Appendix A, GDC 19~~ (Ref. ~~1~~). the UFSAR

DB2 5

~~The Remote Shutdown System is considered an important contributor to reducing the risk of accidents; as such, it has been retained in the Technical Specifications (TS) as indicated in the NRC Policy Statement.~~ TA1

satisfies Criterion 4 of the NRC Policy Statement

LCO

The Remote Shutdown System LCO provides the requirements for the OPERABILITY of the instrumentation and controls necessary to place and maintain the plant in MODE 3 from ~~a~~ location other than the control room. The instrumentation and controls ~~typically~~ required are listed in ~~Table 3.3.3.2-1 in the accompanying LCO.~~

PA1 5 PA1

10 CFR 50.36(c)(2)(ii)(Ref.3)

X2

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PA2

X1

X2

~~Reviewer's Note: For channels that fulfill GDC 19 requirements, the number of OPERABLE channels required depends upon the plant's licensing basis as described in the NRC plant specific Safety Evaluation Report (SER). Generally, two divisions are required to be OPERABLE. However, only one channel per given Function is required if the plant has justified such a design and the NRC SER has accepted the justification.~~

1 J

the Technical Requirements Manual (Reference 4). In addition, as stated in the Technical Requirements Manual, this portion of the Technical Requirements Manual is considered part of these Bases. Thus, changes to the instrumentation and controls listed in the Technical Requirements Manual are controlled by the Technical Specifications Bases Control Program.

The controls, instrumentation, and transfer switches are those required for:

- Reactor pressure vessel (RPV) pressure control;
- Decay heat removal;
- RPV inventory control; and
- Safety support systems for the above functions, including Emergency service water, ~~component cooling~~ RHR service water, Cresent area unit coolers, and onsite power, including the emergency diesel generators.

PA3 DB1 PA3 PA3

The Remote Shutdown System is OPERABLE if all instrument and control channels needed to support the remote shutdown function are OPERABLE. In some cases, ~~Table 3.3.3.2-1 may indicate that~~ the required information or control capability is available from several alternate sources. In these cases, the Remote Shutdown System is OPERABLE as long as one

(continued)

REVISION ~~F~~ I

BASES

ACTIONS (continued)

As such, a Note has been provided that allows separate Condition entry for each inoperable Remote Shutdown System Function.

A.1

Condition A addresses the situation where one or more required Functions of the Remote Shutdown System is inoperable. This includes any Function listed in ~~Table 3.3.3.2-1~~ Reference 4 1 , as well as the control and transfer switches.

The Required Action is to restore the Function (both divisions, if applicable) to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

B.1

If the Required Action and associated Completion Time of Condition A are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

Insert SR Note X3

SR 3.3.3.2.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying

(continued)

Revision I

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.2.1 (continued)

the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Channel

PA1

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. As specified in the Surveillance, a CHANNEL CHECK is only required for those channels that are normally energized.

The Frequency is based upon plant operating experience that demonstrates channel failure is rare.

SR 3.3.3.2.2

SR 3.3.3.2.2 verifies each required Remote Shutdown System transfer switch and control circuit performs the intended function. This verification is performed from the remote shutdown panel and locally, as appropriate. Operation of the equipment from the remote shutdown panel is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the plant can be placed and maintained in MODE 3 from the remote shutdown panel , auxiliary shutdown panels and the local control stations. ~~However, this Surveillance is not required to be performed only during a plant outage.~~ The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience demonstrates that Remote Shutdown System control channels usually pass the Surveillance when performed at the ~~18~~ 24 month Frequency.

PA1

CLB1

PA1

SR 3.3.3.2.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies the channel responds to measured parameter values with the necessary range and accuracy.

The ~~18~~ 24 month Frequency is based upon operating experience and consistency with the ~~typical industry~~ refueling cycle.

CLB1

PA1

J

(continued)

Revision J

BASES (continued)

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 19.~~ UFSAR, Section 16.6 DB2

2. UFSAR, Section 14.5.10.

3. 10 CFR 50.36 (c)(2)(ii).

4. Technical Requirements Manual, Appendix D.

DB2 X2 J

Revision J

BASES (continued)

APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a design capability to promptly shut down the reactor to MODE 3, including the necessary instrumentation and controls, to maintain the plant in a safe condition in MODE 3.

The criteria governing the design and the specific system requirements of the Remote Shutdown System are located in the UFSAR (Refs. 1 and 2).

The Remote Shutdown System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii)(Ref. 3).

LCO

The Remote Shutdown System LCO provides the requirements for the OPERABILITY of the instrumentation and controls necessary to place and maintain the plant in MODE 3 from locations other than the control room. The instrumentation and controls required are listed in the Technical Requirements Manual (Reference 4). In addition, as stated in the Technical Requirements Manual, this portion of the Technical Requirements Manual is considered part of these Bases. Thus, changes to the instrumentation and controls listed in the Technical Requirements Manual are controlled by the Technical Specifications Bases Control Program.

The controls, instrumentation, and transfer switches are those required for:

- Reactor pressure vessel (RPV) pressure control;
- Decay heat removal;
- RPV inventory control; and
- Safety support systems for the above functions, including Emergency Service water, RHR Service water, cresent area unit coolers and onsite power, including the emergency diesel generators.

The Remote Shutdown System is OPERABLE if all instrument and control channels needed to support the remote shutdown function are OPERABLE. In some cases, the required

(continued)

BASES

ACTIONS (continued)

entry into the Condition. However, the Required Actions for inoperable Remote Shutdown System Functions provide appropriate compensatory measures for separate Functions.

As such, a Note has been provided that allows separate Condition entry for each inoperable Remote Shutdown System Function.

A.1

Condition A addresses the situation where one or more required Functions of the Remote Shutdown System is inoperable. This includes any function listed in Reference 4, as well as the control and transfer switches. | J

The Required Action is to restore the Function (both divisions, if applicable) to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

B.1

If the Required Action and associated Completion Time of Condition A are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. The 6 hour testing allowance is acceptable since it does not significantly reduce the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.2.2 (continued)

Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience demonstrates that Remote Shutdown System control channels usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.3.2.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies the channel responds to measured parameter values with the necessary range and accuracy.

The 24 month Frequency is based upon operating experience and consistency with the refueling cycle.

REFERENCES

1. UFSAR, Section 16.6.
2. UFSAR, Section 14.5.10.
3. 10 CFR 50.36(c)(2)(ii).
4. Technical Requirements Manual, Appendix D.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
[Table 3.2-7 Note 1.b] [L2] B. One Function with ATWS-RPT trip capability not maintained.	B.1	Restore ATWS-RPT trip capability.	72 hours
[Table 3.2-7 Note 1.b L2] C. Both Functions with ATWS-RPT trip capability not maintained.	C.1	Restore ATWS-RPT trip capability for one Function.	1 hour
[Table 3.2-7 Note 1.a Note 1.b] [L3] D. Required Action and associated Completion Time not met.	D.1	----NOTE---- Only applicable if inoperable channel is the result of an inoperable RPT breaker ---- Remove the ~~associated~~ affected recirculation pump from service. TA1	6 hours
	OR		
	D.2	Be in MODE 2.	6 hours

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SURVEILLANCE REQUIREMENTS

[Table 3.2-7 Note 2]

-----------------------------------NOTE-----------------------------------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability.

SURVEILLANCE		FREQUENCY
[Table 4.2-7] [M1] SR 3.3.4.2.1 ① PA1	Perform CHANNEL CHECK.	12 hours DB2

(continued)

CLB3

INSERT BKGD

The ATWS-RPT logic consists of two trip systems for the Reactor Vessel Water Level-Low Low (Level 2) trip function and two trip systems for the Reactor Pressure-High trip function. Each trip system associated with the Reactor Vessel Water Level - Low Low (Level 2) Function includes two reactor water level channels while each trip system associated with the Reactor Pressure - High Function includes two reactor pressure channels. Each ATWS trip system is a one-out-of-two logic and both trip systems associated with the same function must trip for the ATWS trip logic to actuate. Therefore, the ATWS trip system logic for each Function is one-out-of-two taken twice.

The two channels in each trip system are powered from a common power supply. For each trip function, the two channels in one trip system are powered independently from the two channels in the other trip system (Divisions 1 and 2). The logic associated with the two trip systems for the Reactor Vessel Water Level - Low Low (Level 2) trip function and the logic associated with the two trip systems for the Reactor Pressure - High trip function are all powered from one common power supply.

credited PA3

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ATWS-RPT is not assumed in the safety analysis. The ATWS-RPT initiates an RPT to aid in preserving the integrity of the fuel cladding following events in which a scram does not, but should, occur. ~~Based on its contribution to the reduction of overall plant risk, however, the instrumentation is included as required by the NRC Policy Statement.~~ 1 J

The ATWS-RPT instrumentation satisfies Criterion 10 CFR 50.36(c)(2)(ii) (Ref. 2) TA4 X1

The OPERABILITY of the ATWS-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.2.4. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Channel OPERABILITY also includes the associated recirculation pump drive motor breakers. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. 1 PA1 MG CLB1

Allowable Values are specified for each ATWS-RPT Function specified in the LCO. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. ~~The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.~~ PA3

PA3 ATWS

Insert ASA DB3

The individual Functions are required to be OPERABLE in MODE 1 to protect against common mode failures of the

(continued)

REVISION F J

(OB3)

INSERT ASA

The trip setpoints are derived from the analytic limits and account for all worst case instrumentation uncertainties as appropriate (e.g., drift, process effects, calibration uncertainties, and severe environmental errors (for channels that must function in harsh environments as defined by 10 CFR 50.49)). The trip setpoints derived in this manner provide adequate protection because all expected uncertainties are accounted for. The Allowable Values are than derived from the trip setpoints by accounting for normal effects that would be seen during periodic surveillance or calibration. These effects are instrumentation uncertainties observed during normal operation (e.g., drift and calibration uncertainties).

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Reactor Vessel Water Level—Low Low, Level 2 1 PA2
(continued)

no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Vessel Water Level—Low Low, Level 2 Allowable Value is chosen so that the system will not be initiated after a Level 3 scram with feedwater still available, and ~~for convenience with the reactor core isolation cooling initiation.~~ PA2 INSERT FUNCTION a DB2 J

b. Reactor ~~Steam Dome~~ Pressure—High PA2

Excessively high RPV pressure may rupture the RCPB. An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This increases neutron flux and THERMAL POWER, which could potentially result in fuel failure and overpressurization. The Reactor ~~Steam Dome~~ Pressure—High Function initiates an RPT for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power generation. For the overpressurization event, the RPT aids in the termination of the ATWS event and, along with the safety/relief valves (SRVs), limits the peak RPV pressure to less than the ASME Section III Code Service Level C limits (1500 psig). PA2 PA2

DB2 INSERT B334.1-1

The Reactor ~~Steam Dome~~ Pressure—High signals are initiated from four pressure transmitters that monitor reactor steam dome pressure. Four channels of Reactor ~~Steam Dome~~ Pressure—High, with two channels in each trip system, are available and are required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor ~~Steam Dome~~ Pressure—High Allowable Value is chosen to provide an adequate margin to the ASME Section III Code Service Level C allowable Reactor Coolant System pressure. PA2 PA2 PA2

ACTIONS

A Note has been provided to modify the ACTIONS related to ATWS-RPT instrumentation channels. Section 1.3, Completion

(continued)

REVISION ~~F~~ J

DB2

INSERT B 3.3.4.1-1

The Allowable Value is dependent on the number of OPERABLE S/RVs. The peak pressure resulting from an ATWS with Main Steam Isolation Valve (MSIV) closure (the limiting transient) is dependent on the power produced during the transient (which is sensitive to the ATWS-RPT Reactor Pressure-High setpoint) and the capability to remove heat from the RPV (which is sensitive to the number of operable S/RVs). The Allowable Value with ≥ 10 S/RVs OPERABLE was derived from the analysis performed in Reference 4. The Allowable Value with < 10 S/RVs OPERABLE was derived from the analysis performed in Reference 5.

DB2

INSERT Function a

also provides an opportunity for the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems to recover water level if feedwater is not available. The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 3). 5

The HPCI, RCIC and ATWS-RPT initiation functions (as described in Table 3.3.5.1-1, Function 3.a; Table 3.3.5.2-1, Function 1; and LCO 3.3.4.1.a including SR 3.3.4.1.4, respectively) describe the reactor vessel water level initiation function as "Low Low (Level 2)." The Allowable Values associated with the HPCI and RCIC initiation function is different from the Allowable Value associated with the ATWS-RPT initiation function as the ATWS function has a separate analog trip unit. Nevertheless, consistent with the nomenclature typically used in design documents, the "Low Low (Level 2)" designation is retained in describing each of these three initiation functions. 5

TA1

INSERT SR 3.3.4.1-2

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

BASES

1 PA1

SURVEILLANCE REQUIREMENTS

SR 3.3.4.~~2~~.3 (continued)

channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of ~~92~~ 184 days is based on the reliability ~~analysis of Reference 2~~ accuracy and low failure rates of these solid-state electronic components. CLB1

1 PA1

SR 3.3.4.~~2~~.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

24 PB4

The Frequency is based upon the assumption of a~~n~~ ~~18~~ 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

1 PA1

SR 3.3.4.~~2~~.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the pump breakers is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a breaker is incapable of operating, the associated instrument channel~~(s)~~ would be inoperable. DB2 15

24 CLB2

The ~~18~~ 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the ~~18~~ 24 month Frequency.

24 CLB2

(continued)

Revision J

DB2 INSERT REF X1

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

3. Drawing 11825-5.01-15D, Rev. D, Reactor Assembly Nuclear Boiler, (GE Drawing 919D690BD).

4. JAF-RPT-MISC-02738 (CHS-96-05), GE letter, FitzPatrick Nuclear Power Plant ATWS Analysis For Recirculation Pump Trip Setpoint Changes High Pressure Trip Setpoint Evaluation, May 23, 1996. 1 I 1 J

5. GE-NE-187-59-1191, FitzPatrick Power Uprate Impact Study Engineering Report: Section 9.3.1, Anticipated Transients Without Scram (ATWS) Analyses for the James A. FitzPatrick Nuclear Power Plant, November, 1991.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One Function with ATWS-RPT trip capability not maintained.	B.1	Restore ATWS-RPT trip capability.	72 hours
C. Both Functions with ATWS-RPT trip capability not maintained.	C.1	Restore ATWS-RPT trip capability for one Function.	1 hour
D. Required Action and associated Completion Time not met.	D.1	-------NOTE--------- Only applicable if inoperable channel is the result of an inoperable RPT breaker. -------------------- Remove the affected recirculation pump from service.	6 hours
	<u>OR</u>		
	D.2	Be in MODE 2.	6 hours

(J)

B 3.3 INSTRUMENTATION

B 3.3.4.1 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

BASES

BACKGROUND

The ATWS-RPT System initiates an RPT, adding negative reactivity, following events in which a scram does not (but should) occur, to lessen the effects of an ATWS event. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area as core flow decreases. When Reactor Vessel Water Level – Low Low (Level 2) or Reactor Pressure – High setpoint is reached, the recirculation pump motor generator (MG) drive motor breakers trip.

The ATWS-RPT System (Ref. 1) includes sensors, logic circuits, relays, and switches that are necessary to cause initiation of an RPT. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an ATWS-RPT signal to the trip logic.

The ATWS-RPT logic consists of two trip systems for the Reactor Vessel Water Level – Low Low (Level 2) trip function and two trip systems for the Reactor Pressure – High trip function. Each trip system associated with the Reactor Vessel Water Level – Low Low (Level 2) Function includes two reactor water level channels while each trip system associated with the Reactor Pressure – High Function includes two reactor pressure channels. Each ATWS trip system is a one-out-of-two logic and both trip systems associated with the same function must trip for the ATWS trip logic to actuate. Therefore, the ATWS trip system logic for each Function is one-out-of-two taken twice.

The two channels in each trip system are powered from a common power supply. For each trip function, the two channels in one trip system are powered independently from the two channels in the other trip system. (Divisions 1 and 2). The logic associated with the two trip systems for the Reactor Vessel Water – Low Low (Level 2) trip function and the logic associated with the two trip systems for the Reactor Pressure – High trip function are all powered from one common power supply.

(continued)

BASES

BACKGROUND (continued)

There is one drive motor breaker provided for each of the recirculation pump MGs for a total of two breakers. The output of each trip function logic is provided to both recirculation pump MG drive motor breakers.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ATWS-RPT is not credited in the safety analysis. The ATWS-RPT initiates an RPT to aid in preserving the integrity of the fuel cladding following events in which a scram does not, but should, occur. ATWS-RPT instrumentation satisfies Criterion 4 of 10 CFR 50.36 (c) (2) (ii) (Ref. 2).

The OPERABILITY of the ATWS-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in both trip systems, with their setpoints within the specified Allowable Value of SR 3.3.4.1.4. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Channel OPERABILITY also includes the associated recirculation pump MG drive motor breakers.

Allowable Values are specified for each ATWS-RPT Function specified in the LCO. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the ATWS analysis. The trip setpoints are derived from the analytical limits and account for all worst case instrumentation uncertainties as appropriate (e.g., drift, process effects, calibration uncertainties, and severe environmental errors (for channels that must function in harsh environments as defined by 10 CFR 50.49)). The trip setpoints derived in this manner provide adequate protection because all expected uncertainties are accounted for. The

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Allowable Values are then derived from the trip setpoints by accounting for normal effects that would be seen during periodic surveillance or calibration. These effects are instrumentation uncertainties observed during normal operation (e.g., drift and calibration uncertainties). 1

The individual Functions are required to be OPERABLE in MODE 1 to protect against common mode failures of the Reactor Protection System by providing a diverse trip to mitigate the consequences of a postulated ATWS event. The Reactor Pressure - High and Reactor Vessel Water Level - Low Low (Level 2) Functions are required to be OPERABLE in MODE 1, since the reactor is producing significant power and the recirculation system could be at high flow. During this MODE, the potential exists for pressure increases or low water level, assuming an ATWS event. In MODE 2, the reactor is at low power and the recirculation system is at low flow; thus, the potential is low for a pressure increase or low water level, assuming an ATWS event. Therefore, the ATWS-RPT is not necessary. In MODES 3 and 4, the reactor is shut down with all control rods inserted; thus, an ATWS event is not significant and the possibility of a significant pressure increase or low water level is negligible. In MODE 5, the one rod out interlock ensures that the reactor remains subcritical; thus, an ATWS event is not significant. In addition, the reactor pressure vessel (RPV) head is not fully tensioned and no pressure transient threat to the reactor coolant pressure boundary (RCPB) exists.

The specific Applicable Safety Analyses and LCO discussions are listed below on a Function by Function basis.

a. Reactor Vessel Water Level - Low Low (Level 2)

Low RPV water level indicates that a reactor scram should have occurred and the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The ATWS-RPT System is initiated at Level 2 to assist in the mitigation of the ATWS event. The resultant reduction of core flow reduces the neutron flux and THERMAL POWER and, therefore, the rate of coolant boiloff.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Reactor Vessel Water Level - Low Low (Level 2) (continued)

Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

Four channels of Reactor Vessel Water Level - Low Low (Level 2), with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Vessel Water Level - Low Low (Level 2) Allowable Value is chosen so that the system will not be initiated after a Level 3 scram with feedwater still available, and also provides an opportunity for the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems to recover water level if feedwater is not available. The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 3). 5

The HPCI, RCIC and ATWS-RPT initiation functions (as described in Table 3.3.5.1-1, Function 3.a; Table 3.3.5.2-1, Function 1; and LCO 3.3.4.1.a including SR 3.3.4.1.4, respectively) describe the reactor vessel water level initiation function as "Low Low (Level 2)." The Allowable Values associated with the HPCI and RCIC initiation function is different from the Allowable Value associated with the ATWS-RPT initiation function as the ATWS function has a separate analog trip unit. Nevertheless, consistent with the nomenclature typically used in design documents, the "Low Low (Level 2)" designation is retained in describing each of these three initiation functions. 5

b. Reactor Pressure - High

Excessively high RPV pressure may rupture the RCPB. An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This increases neutron flux and THERMAL POWER, which could potentially result in fuel failure and overpressurization. The Reactor Pressure - High Function initiates an RPT for transients

(continued)

BASES

ACTIONS (continued)

A.1 and A.2

With one or more channels inoperable, but with ATWS-RPT capability for each Function maintained (refer to Required Action B.1 Bases), the ATWS-RPT System is capable of performing the intended function. However, the reliability and redundancy of the ATWS-RPT instrumentation is reduced, such that a single failure in the same trip system could result in the inability of the ATWS-RPT System to perform the intended function. Therefore, only a limited time is allowed to restore the inoperable channels to OPERABLE status. Because of the diversity of sensors available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting both Functions, and the low probability of an event requiring the initiation of ATWS-RPT, 14 days is provided to restore the inoperable channel (Required Action A.1). Alternately, the inoperable channel may be placed in trip (Required Action A.2), since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As noted, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable breaker, since this may not adequately compensate for the inoperable breaker (e.g., the breaker may be inoperable such that it will not open). If it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel would result in an RPT), or if the inoperable channel is the result of an inoperable breaker, Condition D must be entered and its Required Actions taken.

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining ATWS-RPT trip capability. A Function is considered to be maintaining ATWS-RPT trip capability when sufficient channels are OPERABLE or in trip such that the ATWS-RPT System will generate a trip signal from the given Function on a valid signal, and both recirculation pumps can be tripped. This requires one channel of the Function in each trip system to each be OPERABLE or in trip, and the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into the associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 6) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the recirculation pumps will trip when necessary.

SR 3.3.4.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Channel agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the required channels of this LCO.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 6.

SR 3.3.4.1.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in SR 3.3.4.1.4. If the trip setting is discovered to be less conservative than the setting accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 184 days is based on the reliability, accuracy, and low failure rates of these solid-state electronic components.

SR 3.3.4.1.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1.4 (continued)

range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency 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 3.3.4.1.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the pump breakers is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a breaker is incapable of operating, the associated instrument channels would be inoperable.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Figure 7.4-9 Reactor Recirculation System (FCD).

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

3. Drawing 11825-5.01-15D, Rev. D, Reactor Assembly Nuclear Boiler, (GE Drawing 919D690BD).

4. JAF-RPT-MISC-02738 (CHS-96-05), GE letter, FitzPatrick Nuclear Power Plant ATWS Analysis For Recirculation Pump Trip Setpoint Changes High Pressure Trip Setpoint Evaluation, May 23, 1996.

(continued)

A1

Table 3.3.5.1 Emergency Core Cooling System Instrumentation

ITS Function 2.h

ITS Functions: 1.a, 2.a, 2.e, 3.a, 3.c, 4.a, 4.c, 5.a, 5.c

JAFNPP

TABLE 4.2-2

CORE AND CONTAINMENT COOLING SYSTEM INSTRUMENTATION TEST AND CALIBRATION REQUIREMENTS

~~Instrument~~ Function Channel	Channel Instrument Functional Test [3.3.5.1.2]	Channel Calibration Frequency [SR 3.3.5.1.3, SR 3.3.5.1.4, SR 3.3.5.1.5]	Channel ~~Instrument~~ Check (Note 4) [SR 3.3.5.1.1]
1) Reactor Water Level	2- Q ~~(Note 5)~~ A5	4- SA / R ~~(Note 15)~~ 5	1- ~~D~~ 12 hours M3
2a) Drywell Pressure ~~(non-ATTS)~~ A14	Q A13	3- Q	~~NA~~
2b) Drywell Pressure ~~(ATTS)~~	2- Q ~~(Note 5)~~	4- SA / R ~~(Note 15)~~ 5	1- ~~D~~
3a) Reactor Pressure (non-ATTS)	Q	Q	NA
3b) Reactor Pressure ~~(ATTS)~~ A14	2 Q ~~(Note 5)~~ A5	4- SA / R ~~(Note 15)~~ 5	1- D 12 hours M3
4) Auto Sequencing Timers	NA	5- R	NA
5) ADS - LPCI or CS Pump Disch.	Q A13	3- Q	~~NA~~
6) HPCI & RCIC Suction Source Levels	Q	3- Q	~~NA~~
7) 4kV Emergency Bus Under-Voltage (Loss-of-Voltage, Degraded Voltage LOCA and non-LOCA) Relays and Timers.	R	R	NA

ITS Functions 1.b, 2.b, 3.b

see ITS 3.3.6.1

ITS Functions 1.c, 2.c, 2.d

ITS Functions 1.d, 2.f, 4.b, 5.b

see ITS 3.3.8.1

~~NOTE: See notes following Table 4.2-5.~~

ITS Functions 3.d, 3.e

ITS Functions 4.d, 4.e, 5.d, 5.e

M2 add SR 3.3.5.1.3, SR 3.3.5.1.6 for Functions 1.f, 3.g

add SR 3.3.5.1.5, SR 3.3.5.1.6 for Functions 1.e, 2.g, 3.f M2

J

DISCUSSION OF CHANGES
ITS: 3.3.5.1 - ECCS INSTRUMENTATION

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specification (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 A Note has been added at the start of the CTS Table 3.2-2 Actions Table ("Separate Condition entry is allowed for each channel.") to provide more explicit instructions for proper application of the Actions for Technical Specification compliance (ITS 3.3.5.1 ACTIONS Note). In conjunction with the proposed Specification 1.3 "Completion Times," this Note provides direction consistent with the intent of the Required Actions for inoperable ECCS channels, functions, or trip systems. It is intended that each Required Action be applied regardless of it having been applied previously for other inoperable ECCS channels, functions, trip systems or breakers. This clarification is considered administrative.

A3 The proposed format for this Specification includes an ACTION (ACTION A) that directs entry into the appropriate Conditions referenced in Table 3.3.5.1-1 when one or more channels are inoperable. The ACTION has been added since not all Functions have the same ACTIONS. This change represents a presentation preference only and is, therefore, considered administrative.

A4 CTS 3.2.D and 4.2.D provide a cross reference to the Radiological Effluent Technical Specification (Appendix B) for those Radiation Monitoring Systems which provide an Isolation and Initiation Function. Since CTS 3.2.D and 4.2.D do not prescribe any specific requirements and since the changes to the current requirements in Appendix B are discussed in the Discussion of Changes within this submittal, this cross reference has been deleted. This change is considered administrative since it simply eliminates a cross-reference. This change is consistent with NUREG-1433, Revision 1.

A5 CTS Table 4.2-2, Note 5 states that "This instrumentation is exempt from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel." The ITS definition of Channel Functional Test (CFT) defines a CFT as "the injection of a simulated or actual signal into the channel as close to the sensor as practicable". Therefore, the test defined by CTS Table

ADMINISTRATIVE CHANGES

A5 (continued)

4.2-2 Note 5 is consistent with the ITS definition of a CFT, and the Note can be deleted. Since this represents no new or different requirements it is considered administrative, and is consistent with NUREG-1433, Revision 1.

A6 CTS Table 4.2-2, Note 4 states that "instrument checks are not required when these instruments are not required to be operable or are tripped." ITS SR 3.0.1 states that "SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR." The SR also states that "surveillances do not have to be performed on inoperable equipment or variables outside specified limits." These two statements in SR 3.0.1 equate to Table 4.2-2, Note 2, deleting the need to have the note. Its deletion does not cause a new or different requirement from the CTS. Therefore, the change is considered administrative, and is consistent with NUREG-1433, Revision 1.

A7 The column title in CTS Table 3.2-2 (Total Number of Instrumentation Channels Provided by Design for Both Trip Systems) is proposed to be changed to a per Function basis in ITS 3.3.5.1 rather than the current per Trip System basis. Therefore, except as otherwise noted, the number of channels in the proposed column will be changed to identify the number of channels associated with the new ITS 3.3.5.1 Function. This new categorization is used for all ECCS instrumentation, except the ADS instrumentation. For the ADS instrumentation, each of the two trip systems is listed separately in the Table 3.3.5.1-1 (proposed Functions 4 and 5), thus, the channels per Function do not change. This is considered to be an administrative change.

A8 The details in the CTS Table 3.2-2 “Total Number of Instrumentation Channels Provided by Design for Both Trip Systems” column identifying which systems are supported by the CTS Trip Functions have been deleted (e.g., Core Spray and RHR). ITS Table 3.3.5.1-1 is arranged to identify each Function providing support to a specific ECCS System. Therefore, all the Trip Functions in CTS Table 3.2-2 providing a support Function to the Core Spray System, Low Pressure Injection System (LPCI), High Pressure Coolant Injection (HPCI) System and the Automatic Depressurization System (ADS) Trip System A and B are now associated with the specific System in ITS Table 3.3.5.1-1, thus it is not necessary to identify this cross reference to each system. This change

ADMINISTRATIVE CHANGES

A8 (continued)

in format does not alter any technical requirements, and is therefore, considered administrative. This change is consistent with NUREG-1433, Revision 1.

A9 CTS Table 3.2-2 Item 12 identifies specific start timer setpoints for the "1st Pump" and "2nd Pump" for RHR (LPCI) Loops A and B. In ITS Table 3.3.5.1-1 Function 2.f, the specific LPCI pumps (e.g., A, D) are identified and are associated along with the appropriate Allowable Values. This change in format does not alter any technical requirements and is therefore considered administrative. This change is consistent with NUREG-1433, Revision 1.

A10 CTS 3.2.B requires the Core and Containment Cooling System instrumentation to be Operable whenever the system(s) it initiates or controls are required to be operable as specified in CTS 3.5. CTS 3.5.A.1 and CTS 3.5.A.2 require the Core Spray (CS) and Low Pressure Coolant Injection (LPCI) Systems to be Operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition (ITS MODES 1, 2 and 3). CTS 3.5.C.1 requires the High Pressure Injection System (HPCI) to be Operable whenever the reactor pressure is > 150 psig and reactor coolant temperature is greater than 212°F and irradiated fuel is in the reactor vessel (MODE 1, and MODES 2 and 3 when reactor pressure is > 150 psig), and CTS 3.5.D.1 requires Automatic Depressurization System (ADS) to be Operable whenever the reactor pressure is greater than 100 psig and irradiated fuel is in the reactor vessel. In addition, CTS 3.5.F provides specific Applicability requirements during cold conditions and refueling operations. During a cold condition, two Emergency Core Cooling subsystems are required to be Operable whenever irradiated fuel is in the reactor and work is being performed with the potential for draining the reactor vessel and only one subsystem is required when there are no operations with the potential of draining the vessel. During refueling operations, when the cavity is flooded and the spent fuel pool gates are removed and the water level above the fuel is in accordance with CTS 3.10.C (> 33 feet in the fuel storage pool), no Emergency Core Cooling low pressure subsystems are required.

The CTS 3.5 Applicability during MODES 1, 2 and 3 is consistent with the ITS 3.5.1 (ECCS – Operating) Applicability except for the requirements of ADS. As described in the Discussion of Changes for ITS 3.5.1 the Applicability of ADS has been changed to be consistent with the Applicability of HPCI. The changes to the Applicability requirements

ADMINISTRATIVE CHANGES

A10 (continued)

during cold shutdown (MODE 4) and refueling operations (MODE 5) are discussed in the Discussion of Changes for ITS 3.5.2 (ECCS - Shutdown). The requirements for Operability during MODE 4 are consistent with the current requirements. However, in MODE 5 the minimum water level when no ECCS subsystems are required has been increased to 22 feet 2 inches above the reactor pressure vessel flange. This change is more restrictive than CTS 3.5.F Applicability and is discussed in M3 of the Discussion of Changes for ITS 3.5.2.

The proposed Applicability for ITS 3.3.5.1 is indicated in Table 3.3.5.1-1 for each Function and is consistent with the new Applicability in ITS 3.5.1 and 3.5.2. Since the modifications to the Applicability are adequately discussed in the Discussion of Changes for ITS 3.5.1 and 3.5.3, the corresponding changes to the ECCS instrumentation Applicability are considered administrative since the CTS clearly identifies the instrumentation to be provide a support Function to the associated ECCS subsystems. Any additional changes to the Applicability of any specific Function is discussed below.

Footnote (a) to ITS Table 3.3.5.1-1 requires ECCS subsystems to be operable per LCO 3.5.2, ECCS-Shutdown. This ITS requirement is consistent with CTS requirements as described above. Specifically, the CTS 3.2.B requires the Core and Containment Cooling System instrumentation to be Operable whenever the system(s) it initiates or controls are required to be operable as specified in CTS 3.5. Therefore, the footnote reflects the current licensing basis requirements. Accordingly, the incorporation of this ITS footnote is considered an administrative change.

A11 A Note has been added to CTS Table 3.2-2 Action Notes 1.A to clearly identify the Functions which these actions are applicable to. ITS 3.3.5.1 Required Action B.2 Note specifies that Required Action B.2 is only applicable to ITS 3.3.5.1 Functions 3.a and 3.b. This note is simply a clarification of the current requirements and therefore this change is considered administrative, but aids in the Application of the Required Actions. This change is consistent with NUREG-1433, Revision 1.

ADMINISTRATIVE CHANGES

A12 CTS Table 3.2-2 includes a "Trip Level Setting" column. The setting for each Core and Containment Cooling System Initiation and Control Instrumentation Operability Requirements Functions is listed in this column. In the ITS, the ECCS Functions are included in Table 3.3.5.1-1 along with its associated "Allowable Value".

The CTS "trip level settings" are considered the "Allowable Values" as described in the ITS since the instrumentation is considered inoperable if the value is exceeded when either the CTS or the ITS is applicable. A detailed explanation of trip setpoints, allowable values and analytical limits as they relate to instrumentation uncertainties is provided below.

Trip setpoints are those predetermined values of output at which an action is expected to take place. The setpoints are compared to the actual process parameter and when the measured output value of the process parameter exceeds the setpoint in either the increasing or decreasing direction, the associated device (e.g., trip unit) changes state.

The trip setpoints are specified in the setpoint calculations, are derived from the analytical limits, and account for all worst case applicable instrumentation uncertainties (e.g., drift, process effects, calibration uncertainties, and severe environmental effects as appropriate). The trip setpoints derived in this manner provide adequate protection because all expected uncertainties are accounted for in the setpoint calculations.

The setpoints specified in the setpoint calculations are selected to ensure that the actual field trip setpoints do not exceed the ITS Allowable Values (i.e., the CTS "trip level settings") between successive CHANNEL CALIBRATIONS. The CTS "trip settings" and the "ITS Allowable Values" are both the TS limit values that are placed on the actual field setpoints. The Allowable Values are derived from the trip setpoints by accounting for normal effects that would be seen during periodic surveillance or calibration. These effects are instrumentation uncertainties observed during normal operation (e.g., drift and calibration uncertainties). Accordingly, the ITS Allowable Values include all applicable instrument channel and measurement uncertainties. A channel is inoperable if its actual field trip setpoint is not within its required ITS Allowable Value.

The analytical limits are derived from the limiting values of the process parameters obtained from the safety analysis or other appropriate documents.

ADMINISTRATIVE CHANGES

A12 (continued)

These "Trip Level Settings" or "Allowable Values" have been established consistent with the NYPA Engineering Standards Manual, IES-3A, "Instrument Loop Accuracy and Setpoint Calculation Methodology." The methodology used to determine the "Allowable Values" are consistent with the methodology discussed in ISA-S67.04-1994, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." This change revises the terminology used in the CTS from "Trip Level Setting" to "Allowable Values". Since the instrumentation will be declared inoperable at the same numerical value, this change is considered administrative. Any changes to any "Trip Level Setting" in the CTS will be discussed below. This change is consistent with NUREG-1433, Revision 1.

A13 The explicit requirement to perform a quarterly Functional Test of the Drywell Pressure (non-ATTS), ADS - LPCI or CS Pump Discharge, and HPCI Suction Source Levels instrument channels in CTS Table 4.2-2 is being deleted. CTS Table 4.2-2 and ITS SR 3.3.5.1.3 require a CHANNEL CALIBRATION at the same Frequency, therefore this explicit requirement to perform a quarterly CHANNEL FUNCTIONAL TEST is not required since the ITS definition of CHANNEL CALIBRATION fulfills all the requirements of the CHANNEL FUNCTIONAL TEST. This change is considered administrative since the existing requirements will be fulfilled by performing a CHANNEL CALIBRATION every 92 days.

A14 CTS Table 4.2-2 divides the Surveillance Requirements for Drywell Pressure and Reactor Pressure Functions as non-Analog Transmitter Trip System (ATTS) components and ATTS components. ITS 3.3.5.1 does not specify this explicitly in Table 3.3.5.1-1. Each of the Functions are listed separately along with the associated Surveillance Requirements. Since the required surveillances for non-ATTS and ATTS in the CTS are included in the ITS this change is considered administrative since no technical requirements have been changed. This change is consistent with NUREG-1433, Revision 1.

A15 CTS Table Note 4.2-2 Note 9 requires the calibration of the time delay relays and timers necessary for proper functioning of the trip systems to be performed during the Logic System Functional Test (LFST) which is currently required to be performed every 6 months. This explicit requirement to calibrate the timers during the LSFT has been deleted. The LSFT has been extended to 24 months as justified in L5. The calibration Frequencies of the timers and relays as identified in CTS Table 4.2-2 for the auto sequencing timers is 24 months. The Table does

ADMINISTRATIVE CHANGES

A15 (continued)

not specify a calibration frequency for the ADS auto blowdown timers. Therefore, JAFNPP has interpreted the calibration requirements for all timers to be every 6 months. The calibration Frequency of all timers have been evaluated in accordance with L1 and have been extended from every 6 months to 24 months. Since the calibration and LFST Frequency of all timers have been extended to 24 months, and since all the required timers are explicitly identified as requiring a CALIBRATION and an LFST in ITS Table 3.3.5.1-1 every 24 months, the explicit requirement to perform a calibration of timers during the LFST is not needed. Since the justifications for the Frequency extensions are justified in L1 and L5, this change is considered administrative. This change in format is consistent with NUREG-1433, Revision 1.

A16 CTS Table 3.2-2 Item 17 specifies that the Condensate Storage Tank Low Level setting must be ≥ 59.5 inches above the tank bottom. ITS Table 3.3.5.1-1 Function 3.d does not specify the reference point since it is implied by the associated name of the Function (Condensate Storage Tank Level), however, the current setting is maintained as the Allowable Value (see A12). In addition, CTS Table 3.2-2 Item 18 specifies that the Suppression Pool High Level setting must be ≤ 6 inches above normal level. The normal Suppression Pool Water Level is specified in CTS 3.7.A.1 as being from 13.88 to 14.00 feet. ITS Table 3.3.5.1-1 specifies the Allowable Value to be ≤ 14.5 feet. These changes are considered administrative since there is no technical change in the current requirement. This change is consistent with the format of NUREG-1433, Revision 1.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS Table 3.2-2 Item No. 17 (Condensate Storage Tank Level – Low) requires two channels to be Operable. For the same Function in the ITS (ITS 3.3.5.1-1 Function 3.d) the required number of channels has been increased to 4 channels. The JAFNPP design includes two condensate storage tanks. Both tanks provide suction to the High Pressure Coolant Injection Pump and each tank is instrumented with two channels of Condensate Storage Tank Level – Low. At least one channel in each tank must indicate low water level for the automatic transfer logic to function to initiate the transfer of the suction source from the CSTs to the suppression pool. Therefore to ensure that no single instrument failure can preclude HPCI swap to the suppression pool source four channels of Condensate Storage Tank Level – Low are proposed to be included in the ITS. The addition of new requirements constitutes a more restrictive change.

TECHNICAL CHANGES - MORE RESTRICTIVE

M2 Five additional Functions are proposed to be added to the requirements in CTS Tables 3.2-2 and 4.2-2 to ensure the Core Spray, Low Pressure Coolant Injection (LPCI), and High Pressure Coolant Injection (HPCI) System minimum flow control valves operate as required. Appropriate Actions and Surveillance Requirements have also been added. This instrumentation ensures each minimum flow control valve operates properly. This will ensure each Emergency Core Cooling System pump is Operable and will function as designed during a design basis accident. The addition of new requirements constitutes a more restrictive change. The following Functions have been added as a result of this change:

1.e Core Spray Pump Discharge Flow – Low (Bypass)
1.f Core Spray Pump Discharge Pressure – High (Bypass)
2.g Low Pressure Coolant Injection Pump Discharge Flow – Low (Bypass)
3.f High Pressure Coolant Injection Pump Discharge Flow – Low (Bypass)
3.g High Pressure Coolant Injection Pump Discharge Pressure – High (Bypass)

Functions 1.f and 3.g will be calibrated every 3 months (SR 3.3.5.1.3), Functions 1.e, 2.g, and 3.f will be calibrated every 24 months (SR 3.3.5.1.5), and each Function will be tested every 24 months in accordance with the Logic System Functional Test (SR 3.3.5.1.6). The calibration Frequency is consistent with the setpoint calculation methodology. 5

M3 CTS Table 4.2-2 presently contain daily requirements for performing instrument checks on reactor water level, drywell pressure, and reactor pressure instrumentation. ITS SR 3.3.5.1.1 requires that these Channel Checks be performed every 12 hours. Performing these checks on a more frequent basis adds to the ability to verify that the channels are operable, and therefore, does not represent a change that could affect safety. The channel check ensures once every 12 hours that a gross failure of instrumentation has not occurred. Since the change is requiring a surveillance to be performed on a more frequent basis, the change is considered more restrictive. The proposed change is consistent with NUREG-1433, Revision 1.

M4 Not Used.

M5 Not Used.

Table 3.3.5.1-1 (page 1 of 6)
Emergency Core Cooling System Instrumentation

DB11

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
[T. 3.2-2(2)] [T. 4.2-2(1)] a. Reactor Vessel Water Level — Low Low ~~Low~~, Level 1 (C) (D) (PA1)	1,2,3, 4(a), 5(a)	[4](b) DB6	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.[3] 4 (CLB3) SR 3.3.5.1.5 SR 3.3.5.1.6 ~~SR 3.3.5.1.7~~ (CLB1)	≥ [-113] 18 inches
[T. 3.2-2(8)] [T. 4.2-2(2b)] b. Drywell Pressure — High	1,2,3	[4](b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.[3] 4 (CLB3) SR 3.3.5.1.5 SR 3.3.5.1.6 ~~SR 3.3.5.1.7~~ (CLB1)	≤ [1.92] 2.7 psig
[T. 3.2-2(9)] [T. 4.2-2(3b)] [L6] B c. Reactor ~~Steam Dome~~ Pressure — Low (Injection Permissive) (PA1)	1,2,3	[4]	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.[3] 4 (CLB3) SR 3.3.5.1.5 SR 3.3.5.1.6 ~~SR 3.3.5.1.7~~ (CLB1)	≥ [illegible] 410 psig and ≤ [illegible] 490 psig
	4(a), 5(a)	[4]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.[3] 4 (CLB3) SR 3.3.5.1.5 SR 3.3.5.1.6 ~~SR 3.3.5.1.7~~ (CLB1)	≥ [illegible] 410 psig and ≤ [500] 490 psig
Insert Function 1.d (DB2) [M2] ~~d~~ e. Core Spray Pump Discharge Flow — Low (Bypass) (DB2)	1,2,3, 4(a), 5(a)	[2] [1 per pump]	E	~~SR 3.3.5.1.1~~ ~~SR 3.3.5.1.2~~ ~~SR 3.3.5.1.3~~ (DB7) SR 3.3.5.1.6	≥ [ ] 510 gpm and ≤ [ ] 980 gpm (DB3) [J]
Insert Function 1.f (DB1) [M2] ~~e. Manual Initiation~~	~~1,2,3, 4(a), 5(a)~~	~~[2] [1 per subsystem]~~	~~C~~	~~SR 3.3.5.1.6~~	~~NA~~ (DB4)
2. Low Pressure Coolant Injection (LPCI) System					
[T 3.2-2 (2)] [T 4.2-2 (1)] a. Reactor Vessel Water Level — Low Low ~~Low~~, Level 1 (C) (D) (PA1)	1,2,3, 4(a), 5(a)	[4](b) DB6	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.[3] 4 (CLB3) SR 3.3.5.1.5 SR 3.3.5.1.6 ~~SR 3.3.5.1.7~~ (CLB1)	≥ [-113] 18 inches

(continued)

[A10] (a) When associated ~~subsystem(s)~~ ECCS (TA1) are required to be OPERABLE ~~.~~ emergency per LCO 3.5.2, ECCS-Shutdown (PA1) [J]

(b) Also required to initiate the associated ~~diesel generator~~ subsystem ~~(DG) and isolate the associated plant service water (PSW) turbine building (T/B) isolation valves~~. (TA1) (DB9)

[T 3.2-2 Item 2 Remarks]

REVISION ~~F~~ J

DB2

INSERT Function 1.d

[L7]	d.	Core Spray Pump Start-Time Delay Relay	1,2,3, $4^{(a)}$, $5^{(a)}$	1 per pump	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 12.34 seconds

DB1

INSERT Function 1.f

f.	Core Spray Pump Discharge Pressure – High (Bypass)	1,2,3, $4^{(a)}$, $5^{(a)}$	1 per pump	E	SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 90 psig and ≤ 110 psig

Table 3.3.5.1-1 (page 2 of 6)
Emergency Core Cooling System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE (DB11)
	2. LPCI System (continued)					
[T.3.2-2(8)] [T 4.2-2(2b)]	b. Drywell Pressure — High	1,2,3	[4](b) DB6	B	SR 3.3.5.1.1 SR 3.3.5.1.2 CLB3 [S]R 3.3.5.1.[3] 4 SR 3.3.5.1.5 SR 3.3.5.1.6 CLB1 ~~SR 3.3.5.1.7~~	≤ [1.92] 2.7 psig
[T.3.2-2(9)] [T.4.2-2(3b)] B [L6]	c. Reactor ~~Steam Dome~~ PA1 Pressure — Low (Injection Permissive)	1,2,3	[4]	C	SR 3.3.5.1.1 SR 3.3.5.1.2 CLB3 [S]R 3.3.5.1.[3] 4 SR 3.3.5.1.5 SR 3.3.5.1.6 CLB1 ~~SR 3.3.5.1.7~~	≥ [390] 410 psig and ≤ [500] 490 psig
		4(a), 5(a)	[4]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 CLB3 [S]R 3.3.5.1.[3] 4 SR 3.3.5.1.5 SR 3.3.5.1.6 CLB1 ~~SR 3.3.5.1.7~~	≥ [390] 410 psig and ≤ [500] 490 psig
[T.3.2-2(24)] [T.4.2-2(3b)] B [L6]	d. Reactor ~~Steam Dome~~ PA1 Pressure — Low (Recirculation Discharge Valve Permissive)	1(c), 2(c), 3(c)	[4]	C	SR 3.3.5.1.1 SR 3.3.5.1.2 CLB3 [S]R 3.3.5.1.[3] 4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ [335] 295 psig
[T.3.2-2(5)] [T.4.2-2(1)] [M6]	e. Reactor Vessel Shroud Level ~~&~~ Level 0 PA1	1,2,3	[2]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 CLB3 [S]R 3.3.5.1.[3] 4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ ~~[-202] inches~~ 1.0 inches
[T 3.2-2(12)] [T 4.2-2(4)] [L7]	f. Low Pressure Coolant Injection Pump Start — Time Delay Relay Pumps A[,B],D Pump[s] C B,	1,2,3, 4(a), 5(a)	[4] (1 per pump) DB6	C	SR 3.3.5.1.5 SR 3.3.5.1.6	~~≥ [9] seconds and ≤ [11] seconds~~ 1.51 ≤ 6.73 seconds ~~[5] seconds~~ DB3

(continued)

ECCS TA1

[A10] (a) When associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2.

[T.3.2-2 Item 2 Remarks] (b) Also required to initiate the associated ~~DG and isolate the associated PSW A/B isolation valves~~ emergency diesel generator subsystem. DB9 B

[L3] (c) With associated recirculation pump discharge valve open.

REVISION ~~F~~ J

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE (DB11)
2. LPCI System (continued)		DB6	subsystem	DB7	
[M2] g. Low Pressure Coolant Injection Pump Discharge Flow — Low (Bypass)	1,2,3, 4(a), 5(a)	[1] [1 per pump]	E	~~SR 3.3.5.1.1~~ ~~SR 3.3.5.1.2~~ SR 3.3.5.1.5 SR 3.3.5.1.6	≥ [ ] gpm (1040) and ≤ [ ] gpm (1665) DB3 5
~~h. Manual Initiation~~	~~1,2,3, 4(a), 5(a)~~	~~[2] [1 per subsystem]~~	~~C~~	~~SR 3.3.5.1.6~~	~~NA~~ DB4
[T.3.2-2(6)] add Insert Function 2.h DB12					
3. High Pressure Coolant Injection (HPCI) System					
[T.3.2-2(1)] [T.4.2-2(1)] a. Reactor Vessel Water Level — Low Low ~~Level 2~~ PA1	1, 2(d), 3(d)	[4]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] 4 (CLB3) SR 3.3.5.1.5 SR 3.3.5.1.6 ~~SR 3.3.5.1.7~~ (CLB1)	≥ ~~[-47]~~ 126.5 inches
[T.3.2-2(8)] [T.4.2-2(2b)] b. Drywell Pressure — High	1, 2(d), 3(d)	[4]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] 4 (CLB3) SR 3.3.5.1.5 SR 3.3.5.1.6 ~~SR 3.3.5.1.7~~ (CLB1)	≤ ~~[1.92]~~ 2.7 psig
[T.3.2-2(3)] [T.4.2-2(1)] c. Reactor Vessel Water Level — High ~~Level 8~~ PA1	1, 2(d), 3(d)	[2]	C	SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] 4 (CLB3) SR 3.3.5.1.5 SR 3.3.5.1.6 ~~SR 3.3.5.1.7~~ (CLB1)	≤ ~~[56.5]~~ 222.5 inches
[T.3.2-2(17)] [T.4.2-2(6)] d. Condensate Storage Tank Level — Low	1, 2(d), 3(d)	~~[2]~~ 4	D	~~[SR 3.3.5.1.1]~~ ~~SR 3.3.5.1.2~~ ~~[SR 3.3.5.1.4]~~ SR 3.3.5.1.3 SR 3.3.5.1.6 DB7	≥ ~~[0]~~ 59.5 inches
[T.3.2-2(18)] [T.4.2-2(6)] e. Suppression Pool Water Level — High	1, 2(d), 3(d)	[2] DB6	D	~~SR 3.3.5.1.1~~ ~~SR 3.3.5.1.2~~ [SR 3.3.5.1.3] ~~SR 3.3.5.1.5~~ SR 3.3.5.1.6	≤ ~~[154] inches~~ 14.5 ft 5

(continued)

ECCS TA1 per LCO 3.5.2

[A10] (a) When the associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2. DB10

[A10] (d) With reactor steam dome pressure > [150] psig.

Table 3.3.5.1-1 (page 4 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System (continued)		DB6		DB7	DB11
[M2] f. High Pressure Coolant Injection Pump Discharge Flow – Low (Bypass)	1, $2^{(d)}$, $3^{(d)}$	[1]	E	~~SR 3.3.5.1.1~~ ~~SR 3.3.5.1.2~~ SR 3.3.5.1.5 SR 3.3.5.1.6	≥ [ ] gpm and ≤ [ ] gpm (475; 800) DB3
~~g. Manual Initiation~~	~~1, $2^{(d)}$, $3^{(d)}$~~	~~[1]~~	~~C~~	~~SR 3.3.5.1.6~~	~~NA~~
Insert Function 3.g DB1 DB4					
4. Automatic Depressurization System (ADS) Trip System A					
[T.3.2-2(2)] [T.4.2-2(1)] a. Reactor Vessel Water Level – Low Low Low, Level 1 PA1	1, $2^{(d)}$, $3^{(d)}$	[2]	F	SR 3.3.5.1.1 SR 3.3.5.1.2 (4) [SR 3.3.5.1.3] CLB3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ [-113] inches (18)
~~b. Drywell Pressure – High~~	~~1, $2^{(d)}$, $3^{(d)}$~~	~~[2]~~	~~F~~	~~SR 3.3.5.1.1 SR 3.3.5.1.2 [SR 3.3.5.1.3] SR 3.3.5.1.5 SR 3.3.5.1.6~~	~~≤ [1.92] psig~~ DB4
[T.3.2-2(13)] [T.4.2-2(4)] b. ~~c.~~ Automatic Depressurization System Initiation Timer	1, $2^{(d)}$, $3^{(d)}$	[1]	G	~~[SR 3.3.5.1.3]~~ SR 3.3.5.1.5 DB8 SR 3.3.5.1.6	≤ [120] seconds (134)
[T.3.2-2(1)] [T.4.2-2(1)] c. ~~d.~~ Reactor Vessel Water Level – Low, Level 3 ~~(Confirmatory)~~ PA1	1, $2^{(d)}$, $3^{(d)}$	[1]	F	SR 3.3.5.1.1 (4) SR 3.3.5.1.2 [SR 3.3.5.1.3] CLB3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ [10] inches (177)
[T.3.2-2(15)] [T.4.2-2(5)] d. ~~e.~~ Core Spray Pump Discharge Pressure – High	1, $2^{(d)}$, $3^{(d)}$	[2] DB6	G	~~SR 3.3.5.1.1~~ ~~SR 3.3.5.1.2~~ SR 3.3.5.1.3 ~~[SR 3.3.5.1.3]~~ ~~SR 3.3.5.1.5~~ SR 3.3.5.1.6 DB7	≥ [137] psig and ≤ [ ] psig (90; 110)

DB4 numbering

(continued)

[A10] (d) With reactor steam dome pressure > [150] psig. DB10

REVISION ~~F~~ J

[M2] 1 (J)

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB7 All Functions which include a switch (rather than transmitter and trip unit) in the required channel design, except for Functions 1.e, 2.g, and 3.f, will be calibrated in accordance with the 92 day CHANNEL CALIBRATION SR 3.3.5.1.3. The only other Surveillance associated with these Functions will be the LSFT (SR 3.3.5.1.6). Therefore all other SRs associated with these Functions have been deleted. For Functions 1.e, 2.g, and 3.f, the Channel Calibration (including the Channel Functional Test) will be on a 24 month Frequency, consistent with (or more frequently than) current practice. A review of maintenance history has been performed and shows that a 24 month Frequency for these Functions is acceptable. In addition, the drift analyses in the setpoint calculations support the 24 month Frequency. J J

DB8 All Functions which include Timers will be calibrated in accordance with SR 3.3.5.1.5 (the 24 month CHANNEL CALIBRATION) and SR 3.3.5.1.6 (the LSFT). The Calibration Frequency is consistent with the results of the setpoint calculation for these channels.

DB9 Footnote (b) has been revised to be consistent with the JAFNPP design and consistent with CTS Table 3.2-2 Item 2 Remarks.

DB10 The bracketed Applicability in Table 3.3.5.1-1 Footnote (d) has been revised consistent with the current design requirements as discussed in A10.

DB11 The brackets have been removed from the Allowable Values and the proper plant specific value has been included in accordance with the setpoint calculation methodology.

DB12 A new Function has been added to Table 3.3.5.1-1 to ensure the Low Pressure Coolant Injection subsystems are not diverted unless adequate core cooling is assured and containment spray is needed. This Function is:

2.h Containment Pressure – High

Appropriate Actions and Surveillance Requirements have also been added. This addition is consistent with the current licensing requirements and is consistent with NEDO-31466 (Technical Specification Screening Criteria Application And Risk Assessment), Supplement 1, February 1990.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 275, Revision 0 have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

DB1 unless otherwise noted

BASES

BACKGROUND

Low Pressure Coolant Injection System (continued)

low enough so that pump overheating may occur, the respective minimum flow return line valve is opened. If flow is above the minimum flow setpoint, the valve is automatically closed to allow the full system flow assumed in the analyses.

PA 2

normally closed

return

The RHR ~~test line~~ suppression pool cooling isolation valve, suppression pool spray isolation valves, and containment spray isolation valves (which are also PCIVs) are also closed on a LPCI initiation signal to allow the full system flow assumed in the accident analyses and maintain primary containment isolated in the event LPCI is not operating.

The LPCI System monitors the pressure in the reactor to ensure that, before an injection valve opens, the reactor pressure has fallen to a value below the LPCI System's maximum design pressure. The variable is monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of the trip units are connected to relays whose contacts ~~are~~ arranged in a one-out-of-two taken twice logic. Additionally, instruments are provided to close the recirculation pump discharge valves to ensure that LPCI flow does not bypass the core when it injects into the recirculation lines. The variable is monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of the trip units are connected to relays whose contacts ~~are arranged in a one-out-of-two taken twice logic~~.

provide input to two trip systems.

Insert LPCI-3

Each trip system is

Insert LPCI-4

When the level is greater than the low level setpoint, LPCI may no longer be required, therefore

Low reactor water level in the shroud is detected by two additional instruments ~~to automatically isolate~~ other modes of RHR (e.g., suppression pool cooling) ~~when LPCI is required~~. Manual overrides for these isolations are provided.

are allowed

Insert LPCI-5

below the low level setpoint

Insert LPCI-6

DB10

High Pressure Coolant Injection System (6) (5)

The HPCI System may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level—Low Low, Level 2 or Drywell Pressure—High. Each of these variables is monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of the trip units are

PA1

although manual initiation requires manipulation of individual pump and valve control switches

1(5)

(continued)

Revision I

DB1 unless otherwise noted

DB8

BASES

BACKGROUND — High Pressure Coolant Injection System (continued)

DB8

connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each Function.

(as indicated by the pressure switch)

DB1 — switch

The HPCI pump discharge flow and pressure are ~~is~~ monitored by a flow ~~transmitter~~ switch and pressure switch, respectively. When the pump is running and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow assumed in the accident analysis.

DB1 DB8

The HPCI test line isolation valve ~~(which is also a PCIV)~~ is closed upon receipt of a HPCI initiation signal to allow the full system flow assumed in the accident analysis~~and maintain primary containment isolated in the event HPCI is not operating~~.

J

DB7

The HPCI System also monitors the water levels in the condensate storage tanks (CSTs) and the suppression pool because these are the two sources of water for HPCI operation. Reactor grade water in the CSTs is the normal source. The CST suction source consists of two CSTs connected in parallel to the HPCI pump suction. Upon receipt of a HPCI initiation signal, the CST suction valve is automatically signaled to open (it is normally in the open position) unless both suppression pool suction valves are open. If the water level in both ~~the~~ CSTs falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. Two level switches are used to detect low water level in each ~~the~~ CST. ~~Either switch~~ One switch associated with each CST can cause the suppression pool suction valves to open and the CST suction valve to close. The suppression pool suction valves also automatically open and the CST suction valve closes if high water level is detected in the suppression pool. Two level switches monitor suppression pool water level. Either switch can cause the suppression pool suction valves to open and the CST suction valve to close. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be full open before the other automatically closes.

The HPCI provides makeup water to the reactor until the reactor vessel water level reaches the Reactor Vessel Water Level—High, Level 8 ~~trip~~ setting, at which time the HPCI turbine trips, which causes the turbine's stop valve ~~and the injection valves~~ to close. The logic is two-out-of-two to provide high reliability of the HPCI System. The HPCI

PA1

(continued)

Revision J

DBI unless otherwise noted

BASES

BACKGROUND

High Pressure Coolant Injection System (continued)

System automatically restarts if a Reactor Vessel Water Level—Low Low, Level 2, signal is subsequently received. PAI

Automatic Depressurization System

, although manual initiation requires manipulation of the handswitches associated with each ADS valve.

The ADS may be initiated by either automatic or manual means. Automatic initiation occurs when signals indicating Reactor Vessel Water Level—Low Low Low, Level 1; ~~Drywell Pressure—High or ADS Bypass Low Water Level Actuation Timer;~~ confirmed Reactor Vessel Water Level—Low, Level 3; and CS or LPCI Pump Discharge Pressure—High are all present and the ADS Initiation Timer has timed out. There are two transmitters ~~each~~ for Reactor Vessel Water Level—Low Low Low, Level 1 ~~and Drywell Pressure—High~~, and one transmitter for confirmed Reactor Vessel Water Level—Low, Level 3 in each of the two ADS trip systems. Each of these transmitters connects to a trip unit, which then drives a relay whose contacts form the initiation logic. DB2 PAI

Each ADS trip system includes a time delay between satisfying the initiation logic and the actuation of the ADS valves. The ADS Initiation Timer time delay setpoint chosen is long enough that the HPCI has sufficient operating time to recover to a level above Level 1, yet not so long that the LPCI and CS Systems are unable to adequately cool the fuel if the HPCI fails to maintain that level. An alarm in the control room is annunciated when either of the timers is timing. Resetting the ADS initiation signals prior to time out of the ADS Initiation Timers resets the ADS Initiation Timers. J

DBI

The ADS also monitors the ~~discharge pressures~~ pumps of the four LPCI pumps and the two CS pumps. Each ADS trip system includes two ~~discharge~~ pressure permissive ~~transmitters~~ switches from ~~both~~ one CS and from two LPCI pumps in the associated Division (i.e., Division 1 CS subsystem A and LPCI ~~subsystems~~ A and C input to ADS trip system A, and Division 2 CS subsystem B and LPCI ~~subsystems~~ B and D input to ADS trip system B). The signals are used as a permissive for ADS actuation, indicating that there is a source of core coolant available once the ADS has depressurized the vessel. Any one of the six low pressure pumps is sufficient to permit automatic depressurization.

CLB1

The switches associated with one ADS trip system also provide signals to the other ADS trip system, but these signals are not required for the other ADS trip system to be considered OPERABLE.

(continued)

Revision J

TA 2

INSERT ASA-1

Table 3.3.5.1-1 is modified by two footnotes. Footnote (a) is added to clarify that the associated functions are required to be OPERABLE in MODES 4 and 5 only when their supported ECCS are required to be OPERABLE per LCO 3.5.2, "ECCS – Shutdown." 1 J

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined, accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

DB1 or other appropriate documents

Insert A3A-2 DB3

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions that may require ECCS (or DG) initiation to mitigate the consequences of a design basis transient or accident. To ensure reliable ECCS and DG function, a combination of Functions is required to provide primary and secondary initiation signals. E PA1

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Core Spray and Low Pressure Coolant Injection Systems PA1

1.a, 2.a. Reactor Vessel Water Level—Low Low Low, Level 1

DB1 The EDGs are initiated from Function 1.a and 2.a

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The low pressure ECCS and associated DGs are initiated at Level 1 to ensure that core spray and flooding functions are available to prevent or minimize fuel damage. The Reactor Vessel Water Level—Low Low Low, Level 1 is one of the Functions assumed to be OPERABLE and capable of initiating the ECCS during the transients analyzed in References 2 and 3. In addition, the Reactor Vessel Water Level—Low Low Low, Level 1 Function is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. PA1 E DB4 1 1,2,and 4 5 DB4 1

(continued)

Revision J

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a, 2.a. Reactor Vessel Water Level—Low Low Low, Level 1 (continued)

Reactor Vessel Water Level—Low Low Low, Level 1 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level—Low Low Low, Level 1 Allowable Value is chosen to allow time for the low pressure core flooding systems to activate and provide adequate cooling. Insert Functions 1.a, 2.a-1

Thus, four channels of the CS and LPCI

Four channels of Reactor Vessel Water Level—Low Low Low, Level 1 Function are only required to be OPERABLE when the ECCS or DG(s) are required to be OPERABLE to ensure that no single instrument failure can preclude ECCS and DG initiation. Refer to LCO 3.5.1 and LCO 3.5.2, "ECCS—Shutdown," for Applicability Bases for the low pressure ECCS subsystems; LCO 3.8.1, "AC Sources—Operating"; and LCO 3.8.2, "AC Sources—Shutdown," for Applicability Bases for the DGs.

INSERT FUNCTIONS 1.a, 2a-2

1.b, 2.b. Drywell Pressure—High

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS and associated DGs are initiated upon receipt of the Drywell Pressure—High Function in order to minimize the possibility of fuel damage. The Drywell Pressure—High Function, along with the Reactor Water Level—Low Low Low, Level 1 Function, is directly assumed in the analysis of the recirculation line break (Ref. 4). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

The EDGs are initiated from Function 1.b and 2.b.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

The Drywell Pressure—High Function is required to be OPERABLE when the ECCS or DG(s) are required to be OPERABLE in conjunction with times when the primary containment is

(continued)

REVISION I

DB1 INSERT Function 1.a, 2.a-1

The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 6). 5

TA2 INSERT Function 1a, 2.a-2

Per Footnote (a) to Table 3.3.5.1-1, this ECCS Function is only required to be OPERABLE in MODES 4 and 5 whenever the associated ECCS is required to be OPERABLE per LCO 3.5.2.

DB5

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~1.e, 2.h. Manual Initiation~~

~~The Manual Initiation push button channels introduce signals into the appropriate ECCS logic to provide manual initiation capability and are redundant to the automatic protective instrumentation. There is one push button for each of the CS and LPCI subsystems (i.e., two for CS and two for LPCI).~~

~~The Manual Initiation Function is not assumed in any accident or transient analyses in the FSAR. However, the Function is retained for overall redundancy and diversity of the low pressure ECCS function as required by the NRC in the plant licensing basis.~~

~~There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons. Each channel of the Manual Initiation Function (one channel per subsystem) is only required to be OPERABLE when the associated ECCS is required to be OPERABLE. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.~~

2.d. Reactor ~~Steam Dome~~ Pressure—Low (Recirculation Discharge Valve Permissive) PA1 PA1

Low reactor ~~steam dome~~ pressure signals are used as permissives for recirculation discharge valve closure. This ensures that the LPCI subsystems inject into the proper RPV location assumed in the safety analysis. The Reactor ~~Steam Dome~~ PA1 Pressure—Low is one of the Functions assumed to be OPERABLE and capable of closing the valve during the transients analyzed in References ~~1 and~~ 3. The core cooling PB4 function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains 3 below the limits of 10 CFR 50.46. The Reactor ~~Steam Dome~~ Pressure—Low Function is directly assumed in the analysis of the recirculation line break (Ref. ~~2~~). 1 and 4 5 DB4

The Reactor ~~Steam Dome~~ PA1 Pressure—Low signals are initiated from four pressure transmitters that sense the reactor dome pressure.

The Allowable Value is chosen to ensure that the valves close prior to commencement of LPCI injection flow into the core, as assumed in the safety analysis.

(continued)

Revision I

DBC

<u>INSERT Function 2.e</u>

The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 6).

DB 10

INSERT Function 2.h

2.h Containment Pressure – High

The Containment Pressure – High Function is provided as an isolation of the containment spray mode of RHR on decreasing containment pressure following manual actuation of the system. This isolation ensures excessive depressurization of the containment does not occur due to containment spray actuation. This Function also serves as an interlock permissive to allow the RHR System to be manually aligned from the LPCI mode to the containment spray mode after containment pressure has exceeded the trip setting. The permissive ensures that containment pressure is elevated before the manual transfer is allowed. This ensures that LPCI is available to prevent or minimize fuel damage until such time that the operator determines that containment pressure control is needed. The Containment Pressure – High Function is implicitly assumed in the analysis of LOCAs inside containment (Refs. 1, 2 and 4) since the analysis assumes that containment spray occurs when containment pressure is high.

Containment Pressure – High signals are initiated from four pressure switches that sense drywell pressure. The Containment Pressure – High lower Allowable Value is chosen to ensure isolation of containment spray prior to a negative containment pressure occurring. This maintains margin to the negative design pressure and minimizes operation of the reactor building-to-suppression chamber vacuum breakers, which in turn prevents de-inerting the atmosphere. The upper Allowable Value is chosen to ensure containment spray is not isolated when there may be a need for containment spray. 5

Four channels of the Containment Pressure – High Function are only required to be OPERABLE in MODES 1, 2 and 3. In MODES 4 and 5, containment spray is not assumed to be initiated, and other administrative controls are adequate to control the valves that this Function isolates.

High Pressure Coolant Injection PA2

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

HPCI System

PA1

3.a. Reactor Vessel Water Level—Low Low, Level 2

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the HPCI System is initiated at Level 2 to maintain level above the top of the active fuel. The Reactor Vessel Water Level—Low Low, Level 2 is one of the Functions assumed to be OPERABLE and capable of initiating HPCI during the transients analyzed in References 1 and 3. Additionally, the Reactor Vessel Water Level—Low Low, Level 2 Function associated with HPCI is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Insert Function 3.a(1) PA1 DB1

DB4 to be OPERABLE and capable of initiating HPCI DB1

5 1 and 4 DB4 PA1

Reactor Vessel Water Level—Low Low, Level 2 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level—Low Low, Level 2 Allowable Value is high enough such that for complete loss of feedwater flow, the Reactor Core Isolation Cooling (RCIC) System flow with HPCI assumed to fail will be sufficient to avoid initiation of low pressure ECCS at Reactor Vessel Water Level—Low Low Low, Level 1.

PA1 Insert Function 3.a(2) DB1

Four channels of Reactor Vessel Water Level—Low Low, Level 2 Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for HPCI Applicability Bases.

PA1

DB1 Insert Function 3.b

3.b. Drywell Pressure—High

High pressure in the drywell could indicate a break in the RCPB. The HPCI System is initiated upon receipt of the Drywell Pressure—High Function in order to minimize the possibility of fuel damage. The Drywell Pressure—High Function, along with the Reactor Water Level—Low Low, Level 2 Function, is directly assumed in the analysis of the

PA1 DB1 to be OPERABLE and capable of initiating HPCI

(continued)

Revision I

DBI INSERT Function 3.a (1)

In addition, the Standby Gas Treatment (SGT) System suction valves receive an open signal so that the gland seal exhaust from the HPCI turbine can be treated. Opening of the SGT System suction valves results in automatic starting of the SGT System.

DBI INSERT Function 3.a (2)

The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 6) 1 J

The HPCI, RCIC and ATWS-RPT initiation functions (as described in Table 3.3.5.1-1, Function 3.a; Table 3.3.5.2-1, Function 1; and LCO 3.3.4.1.a including SR 3.3.4.1.4, respectively) describe the reactor vessel water level initiation function as "Low Low (Level 2)." The Allowable Values associated with the HPCI and RCIC initiation function is different from the Allowable Value associated with the ATWS-RPT initiation function as the ATWS function has a separate analog trip unit. Nevertheless, consistent with the nomenclature typically used in design documents, the "Low Low (Level 2)" is retained in describing each of these three initiation functions. 1 J

DBI INSERT Function 3.b

In addition, the SGT System suction valves receive an open signal so that the gland seal exhaust from the HPCI turbine can be treated. Opening of the SGT System suction valves results in automatic starting of SGT.

BASES

1, and

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.b. Drywell Pressure—High (continued) DB4

~~recirculation~~ line break (Ref. 4). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible to be indicative of a LOCA inside primary containment.

Four channels of the Drywell Pressure—High Function are required to be OPERABLE when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for the Applicability Bases for the HPCI System.

3.c. Reactor Vessel Water Level—High, Level 8 PA1

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Level 8 signal is used to trip the HPCI turbine to prevent overflow into the main steam lines (MSLs). The Reactor Vessel Water Level—High, Level 8 Function is not assumed in the accident and transient analyses. It was retained since it is a potentially significant contributor to risk. PA1

Reactor Vessel Water Level—High, Level 8 signals for HPCI are initiated from two level transmitters from the narrow range water level measurement instrumentation. Both Level 8 signals are required in order to ~~close~~ trip the HPCI ~~injection~~ turbine ~~valve~~. This ensures that no single instrument failure can preclude HPCI initiation. The Reactor Vessel Water Level—High, Level 8 Allowable Value is chosen to prevent flow from the HPCI System from overflowing into the MSLs. DB1

PA1

Two channels of Reactor Vessel Water Level—High, Level 8 Function are required to be OPERABLE only when HPCI is required to be OPERABLE. Refer to LCO 3.5.1 ~~and LCO 3.5.2~~ for HPCI Applicability Bases. PA1 PA7

Insert Function 3.c DB1

(continued)

Revision J

DB1

INSERT Function 3.c

The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 6). J

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.e. Suppression Pool Water Level—High (continued)

This Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool.

Suppression Pool Water Level—High signals are initiated from two level switches. The logic is arranged such that either switch can cause the suppression pool suction valves to open and the CST suction valve to close. The Allowable Value for the Suppression Pool Water Level—High Function is chosen to ensure that HPCI will be aligned for suction from the suppression pool before the water level reaches the point at which suppression pool design loads would be exceeded.

Two channels of Suppression Pool Water Level—High Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI swap to suppression pool source. Refer to LCO 3.5.1 for HPCI Applicability Bases.

DB8 , 3.g

DB8 , High Pressure Coolant Injection Pump Discharge Pressure - High (Bypass)

3.f. High Pressure Coolant Injection Pump Discharge Flow—Low (Bypass)

DB1 at reduced flow

DB8 (if the HPCI pump is operating)

The minimum flow instruments are provided to protect the HPCI pump from overheating when the pump is operating ~~and the associated injection valve is not fully open~~. The minimum flow line valve is opened when low flow is sensed, and the valve is automatically closed when the flow rate is adequate to protect the pump. The High Pressure Coolant Injection Pump Discharge Flow—Low Function ~~is~~ are assumed to be OPERABLE and capable of closing the minimum flow valve to ensure that the ECCS flow assumed during the transients and accidents analyzed in References 1, 2, and ~~3~~ 4 are met. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

DB8 discharge

DB8 Pump operation is determined by sensing high pump discharge pressure

DB8 and Pump Discharge Pressure - High

opening the minimum flow valve to protect the pump and

5

J

DB4

DB1 switch

One flow ~~transmitter~~ is used to detect the HPCI System's flow rate. The logic is arranged such that the ~~transmitter~~ causes the minimum flow valve to open. The logic will close the minimum flow valve once the flow closure setpoint is exceeded.

DB8 and one pressure switch is used to detect the HPCI pump discharge pressure.

PA2

DB8 flow switch and pressure switch must activate to

(continued) DB1

Revision J

DB8

INSERT 3.g

The High Pressure Coolant Injection Pump Discharge Pressure - High (Bypass) Allowable Value is less than the pump discharge pressure when the pump is operating in a full flow mode and high enough to avoid any condition that results in a discharge pressure permissive when the HPCI pump is aligned for injection and the pump is not running.

INSERT Function 4.a, 5.a

The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 6).

INSERT Function 4.c, 5.c

The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 6).

DBI all changes unless otherwise noted

BASES

ACTIONS

B.1, B.2, and B.3 (continued) PAI

pressure ECCS and E DGs being concurrently declared inoperable.

such that trip capability is lost

For Required Action B.2, or more redundant automatic HPCI initiation capability is lost if two or more Function 3.a or two Function 3.b channels are inoperable and untripped ~~in the same trip system~~. In this situation (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action B.3 is not appropriate and the HPCI system ~~feature(s) associated with the inoperable, untripped channels~~ must be declared inoperable within 1 hour. As noted (Note 1 to Required Action B.1), Required Action B.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the low pressure ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of initiation capability for 24 hours (as allowed by Required Action B.3) is allowed during MODES 4 and 5. There is no similar Note provided for Required Action B.2 since HPCI instrumentation is not required in MODES 4 and 5; thus, a Note is not necessary.

Notes are also provided (Note 2 to Required Action B.1 and the Note to Required Action B.2) to delineate which Required Action is applicable for each Function that requires entry into Condition B if an associated channel is inoperable. This ensures that the proper loss of initiation capability check is performed. Required Action B.1 (the Required Action for certain inoperable channels in the low pressure ECCS subsystems) is not applicable to Functions 2.e and 2.h, since ese this Functions provides backup to administrative controls ensuring that operators do not divert LPCI flow from injecting into the core when needed, and do not spray the containment unless needed. Thus, a total loss of Function 2.e or 2.h capability for 24 hours is allowed, since the LPCI subsystems remain capable of performing their intended function. J

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action B.1, the Completion Time only begins upon discovery that a redundant feature in the same system (e.g., both CS subsystems) cannot be automatically initiated due to inoperable, untripped channels within the same

(continued)

Revision J

BASES

ACTIONS

B.1, B.2, and B.3 (continued)

Function as described in the paragraph above. For Required Action B.2, the Completion Time only begins upon discovery that the HPCI System cannot be automatically initiated due to two inoperable, untripped channels for the associated Function in the same trip system. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

DB4

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. ~~6~~ 7) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.3. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition H must be entered and its Required Action taken.

DB1 unless otherwise noted

C.1 and C.2

DB6

two Function 1.d channels are inoperable, (c)

such that both trip systems lose initiation capability

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the same Function result in redundant automatic initiation capability being lost for the feature(s). Required Action C.1 features would be those that are initiated by Functions 1.c, 1.d, 2.c, 2.d, and 2.f (i.e., low pressure ECCS). Redundant automatic initiation capability is lost if either (a) two or more Function 1.c channels are inoperable ~~in the same trip system~~, (b) two or more Function 2.c channels are inoperable ~~in the same trip system~~, ~~(c)~~ (d) two or more Function 2.d channels are inoperable ~~in the same trip system~~, or ~~(d)~~ (e) ~~two or more~~ Three Function 2.f channels are inoperable. In this situation (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action C.2 is not appropriate and the feature(s) associated with the inoperable channels must be declared inoperable within 1 hour. Since each

(continued)

Revision J

TA2

INSERT SR 3.3.5.1.2

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. 1 J

CLB3

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.1.~~3~~ 4

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.5.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analyses. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than the setting accounted for in the appropriate setpoint methodology. 1.B4

The Frequency of 92 days is based on the reliability ~~analysis of Reference 5~~. accuracy, and lower failure rates of the associated solid-state electronic Analog Transmitter/Trip System components

CLB3 SR 3.3.5.1.~~4~~ 3 and SR 3.3.5.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of SR 3.3.5.1.~~4~~ 3 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

DB9 24

The Frequency of SR 3.3.5.1.5 is based upon the assumption of ~~an~~ ~~18~~ month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. 1 J

SR 3.3.5.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.1, LCO 3.5.2, LCO 3.8.1, and LCO 3.8.2 overlaps this Surveillance to complete testing of the assumed safety function.

(continued)

Revision J

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1.6 (continued)

The ~~18~~ 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the ~~18~~ 24 month Frequency.

CLB4

~~SR 3.3.5.1.7~~

~~This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in Reference 4.~~

~~ECCS RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. The 18 month Frequency is consistent with the typical industry 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.~~

CLB2

REFERENCES

DB4 PA1

1. ~~U~~FSAR, Section ~~[5.2]~~ 6.5.
2. ~~U~~FSAR, Section ~~[6.3]~~ 14.6.
3. ~~U~~FSAR, ~~Chapter [15]~~ Section 14.5.
4. NEDC-313~~76~~17P, "~~Edwin I. Hatch~~ James A. FitzPatrick Nuclear Power Plant, ~~SAFER/GESTR-LOCA~~, Loss-of-Coolant Accident Analysis," ~~December 1986~~ Revision 2, April 1993. PA6
~~5~~ 7. NEDC-30936(P)-A, "BWR Owners' Group Technical Specification Improvement ~~Analyses~~ Methodology (with Demonstration for BWR ECCS Actuation Instrumentation), Part 2," December 1988. PA6 J

Insert Ref X1 DB1

Revision J

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA4 The "Reviewer's Note" has been deleted.

PA5 ITS 3.3.5.1 Required Action G.1 Note has been deleted since it does not provide any useful guidance. The associated words in the Bases have been deleted, as applicable.

PA6 The quotations used in the Bases References have been removed. The Writer's Guide does not require the use of quotations.

PA7 The reference to LCO 3.5.2 has been deleted since LCO 3.5.2 does not cover the HPCI System. J

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific design. References have been included, as applicable.

DB2 JAFNPP ADS logic does not contain Drywell Pressure – High (ISTS 3.3.5.1 Functions 4.b and 5.b) and ADS Low Water Level Actuation Timer inputs (ISTS 3.3.5.1 Functions 4.g and 5.g). Changes have been made to delete these instruments from the Specification. Therefore, the Bases has been revised to delete the descriptions of theses Functions and renumber other functions as required.

DB3 The description of the setpoint calculation methodology has been revised to reflect the plant specific methodology.

DB4 The proper plant specific references have been provided.

DB5 Bracketed Table 3.3.5.1-1 Functions 1.e (CS Manual Initiation), 2.h (LPCI Manual Initiation), 3.g (HPCI Manual Initiation), 4.h (ADS Trip System A Manual Initiation), and 5.h (ADS Trip System B Manual Initiation) do not apply to the JAFNPP design and therefore are not retained in the ITS. The Bases description has been modified as required to reflect this change.

DB6 An additional Function has been added to Table 3.3.5.1-1 to ensure the Core Spray pump will start within the time constraints required by the ECCS analysis and when required to minimize excess loading on the emergency buses and emergency diesel generators. This Function was added as:

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB6 (continued)

1.d Core Spray Pump Start-Time Delay Relay

The Bases has been revised to reflect this change. Subsequent Functions have been renumbered as required to reflect this change.

DB7 The JAFNPP design includes two condensate storage tanks (CSTs) which provide a source of water to the High Pressure Coolant Injections System. The Bases has been revised to reflect this difference in design.

DB8 Two additional Functions have been added to Table 3.3.5.1-1 to ensure the Core Spray and High Pressure Coolant Injection (HPCI) System minimum flow control valves operate as required. These Functions are:

1.f Core Spray Pump Discharge Pressure – High (Bypass)
3.g High Pressure Coolant (Bypass) Injection Pump Discharge Pressure – High

Appropriate Actions and Surveillance Requirements have also been added. This instrumentation serves to protect the system pumps from overheating when the pump is operating and the associated injection valve is not open. This will ensure the associated Emergency Core Cooling Systems are Operable and will function properly during a design basis accident.

DB9 The 18 month calibration SR Frequency has been extended from 18 months to 24 months consistent with the setpoint calculation methodology and consistent with CTS Table 4.2-2.

DB10 A new Function has been added to Table 3.3.5.1-1 to ensure the Low Pressure Coolant Injection subsystems are not diverted unless containment spray is needed. This Function is:

2.h Containment Pressure – High

Appropriate Actions and Surveillance Requirements have also been added. This addition is consistent with the current licensing requirements and is consistent with NEDO-31466 (Technical Specification Screening Criteria Application And Risk Assessment), Supplement 1, February 1990. The Bases has been modified as required to reflect this change.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.3.5.1 - ECCS INSTRUMENTATION

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 205, Revision 3 have been incorporated into the revised Improved Technical Specifications.

TA2 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 275, Revision 0 have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 NUREG-1433, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995. Subsequent References have been renumbered, as applicable.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	--------NOTE-------- Only applicable for Functions 3.a and 3.b. -------------------- Declare High Pressure Coolant Injection (HPCI) System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	<u>AND</u>		
	B.3	Place channel in trip.	24 hours
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.c, 1.d, 2.c, 2.d, and 2.f. -------------------- Declare supported feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	<u>AND</u>		
	C.2	Restore channel to OPERABLE status.	24 hours

(continued)

Table 3.3.5.1-1 (page 1 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level – Low Low Low (Level 1)	1,2,3, $4^{(a)}$, $5^{(a)}$	$4^{(b)}$	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 18 inches
b. Drywell Pressure – High	1,2,3	$4^{(b)}$	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.7 psig
c. Reactor Pressure – Low (Injection Permissive)	1,2,3	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 410 psig and ≤ 490 psig
	$4^{(a)}$, $5^{(a)}$	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 410 psig and ≤ 490 psig
d. Core Spray Pump Start-Time Delay Relay	1,2,3, $4^{(a)}$, $5^{(a)}$	1 per pump	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 12.34 seconds
e. Core Spray Pump Discharge Flow – Low (Bypass)	1,2,3, $4^{(a)}$, $5^{(a)}$	1 per pump	E	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 510 gpm and ≤ 980 gpm
f. Core Spray Pump Discharge Pressure – High (Bypass)	1,2,3, $4^{(a)}$, $5^{(a)}$	1 per pump	E	SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 90 psig and ≤ 110 psig
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level – Low Low Low (Level 1)	1,2,3, $4^{(a)}$, $5^{(a)}$	$4^{(b)}$	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 18 inches

(continued)

(a) When associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2, ECCS – Shutdown.

(b) Also required to initiate the associated emergency diesel generator subsystem.

Table 3.3.5.1-1 (page 2 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
b. Drywell Pressure – High	1,2,3	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.7 psig
c. Reactor Pressure – Low (Injection Permissive)	1,2,3	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 410 psig and ≤ 490 psig
	4(a), 5(a)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 410 psig and ≤ 490 psig
d. Reactor Pressure – Low (Recirculation Discharge Valve Permissive)	1(c), 2(c), 3(c)	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 295 psig
e. Reactor Vessel Shroud Level (Level 0)	1,2,3	2	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 1.0 inches
f. Low Pressure Coolant Injection Pump Start-Time Delay Relay	1,2,3, 4(a), 5(a)	1 per pump	C	SR 3.3.5.1.5 SR 3.3.5.1.6	
Pumps A, D					≤ 1.51 seconds
Pumps B, C					≤ 6.73 seconds

(continued)

(a) When associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2.

(b) Also required to initiate the associated emergency diesel generator subsystem.

(c) With associated recirculation pump discharge valve open.

| J

Table 3.3.5.1-1 (page 3 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
g. Low Pressure Coolant Injection Pump Discharge Flow – Low (Bypass)	1,2,3, $4^{(a)}$, $5^{(a)}$	1 per subsystem	E	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 1040 gpm and ≤ 1665 gpm
h. Containment Pressure – High	1,2,3	4	B	SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 1 psig and ≤ 2.7 psig
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level – Low Low (Level 2)	1, $2^{(d)}$, $3^{(d)}$	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 126.5 inches
b. Drywell Pressure – High	1, $2^{(d)}$,$3^{(d)}$	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.7 psig
c. Reactor Vessel Water Level – High (Level 8)	1, $2^{(d)}$, $3^{(d)}$	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 222.5 inches
d. Condensate Storage Tank Level – Low	1, $2^{(d)}$, $3^{(d)}$	4	D	SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 59.5 inches
e. Suppression Pool Water Level – High	1, $2^{(d)}$, $3^{(d)}$	2	D	SR 3.3.5.1.3 SR 3.3.5.1.6	≤ 14.5 ft

(continued)

(a) When the associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2.

(d) With reactor steam dome pressure > 150 psig.

Table 3.3.5.1-1 (page 4 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System (continued)					
f. High Pressure Coolant Injection Pump Discharge Flow – Low (Bypass)	1, 2(d), 3(d)	1	E	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 475 gpm and ≤ 800 gpm
g. High Pressure Coolant Injection Pump Discharge Pressure – High (Bypass)	1, 2(d), 3(d)	1	E	SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 25 psig and ≤ 80 psig
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level – Low Low Low (Level 1)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 18 inches
b. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 134 seconds
c. Reactor Vessel Water Level – Low (Level 3)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 177 inches
d. Core Spray Pump Discharge Pressure – High	1, 2(d), 3(d)	2	G	SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 90 psig and ≤ 110 psig

(continued)

(d) With reactor steam dome pressure > 150 psig.

B 3.3 INSTRUMENTATION

B 3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

BASES

BACKGROUND

The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design basis accident or transient.

For most abnormal operational transients and Design Basis Accidents (DBAs), a wide range of dependent and independent parameters are monitored.

The ECCS instrumentation actuates core spray (CS), low pressure coolant injection (LPCI), high pressure coolant injection (HPCI), Automatic Depressurization System (ADS), and the emergency diesel generators (EDGs). The equipment involved with each of these systems is described in the Bases for LCO 3.5.1, "ECCS - Operating" and LCO 3.8.1, "AC Sources- Operating."

<u>Core Spray System</u>

The CS System may be initiated by either automatic or manual means, although manual initiation requires manipulation of individual pump and valve control switches. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low Low (Level 1) or Drywell Pressure - High; or both. Each of these diverse variables is monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of the four trip units associated with each diverse variable are connected to relays whose contacts provide input to two trip systems. Each trip system is arranged in a one-out-of-two taken twice logic for each Function. Each trip system initiates one of two CS pumps and provides an open signal to both injection valves associated with the same CS pump. Once an initiation signal is received by the CS control circuitry, the signal is sealed in until manually reset.

Upon receipt of an initiation signal, if preferred power is available, both CS pumps start after approximately an 11 second time delay. If a CS initiation signal is received when preferred power is not available, the CS pumps start

(continued)

BASES

BACKGROUND

Core Spray System (continued)

after approximately 11 seconds after the bus is energized by the EDGs.

The normally closed CS test line isolation valve, which is also a primary containment isolation valve (PCIV), is closed on a CS initiation signal to allow full system flow assumed in the accident analyses and maintain primary containment isolated in the event CS is not operating.

The CS pump discharge flow and pressure are monitored by a differential pressure indicating switch and a pressure switch, respectively. When the pump is running (as indicated by the pressure switch) and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow assumed in the accident analysis.

The CS System also monitors the pressure in the reactor to ensure that, before the injection valves open, the reactor pressure has fallen to a value below the CS System's maximum design pressure. The variable is monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of the trip units are connected to relays whose contacts provide input to two trip systems. Each trip system is arranged in a one-out-of-two taken twice logic. Each trip system provides an open permissive signal for two CS injection valves in one of the two CS Systems.

Low Pressure Coolant Injection System

The LPCI is an operating mode of the Residual Heat Removal (RHR) System, with two LPCI subsystems. The LPCI subsystems may be initiated by automatic or manual means, although manual initiation requires manipulation of individual pump and valve control switches. Automatic initiation occurs for conditions of Reactor Vessel Water Level – Low Low Low (Level 1); Drywell Pressure – High; or both. Each of these diverse variables is monitored by four redundant transmitters, which, in turn, are connected to four trip units. The outputs of the four trip units associated with

(continued)

BASES

BACKGROUND

Low Pressure Coolant Injection System (continued)

The LPCI System monitors the pressure in the reactor to ensure that, before an injection valve opens, the reactor pressure has fallen to a value below the LPCI System's maximum design pressure. The variable is monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of the trip units are connected to relays whose contacts provide input to two trip systems. Each trip system is arranged in a one-out-of-two taken twice logic. Each trip system provides an open permissive signal to both inboard injection valves and provides an open permissive signal to the associated outboard injection valve. The open permissive signal for the outboard injection valve is maintained for five minutes to ensure the valve fully opens. Additionally, instruments are provided to close the recirculation pump discharge valves to ensure that LPCI flow does not bypass the core when it injects into the recirculation lines. The variable is monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of the trip units are connected to relays whose contacts provide input to two trip systems. Each trip system is arranged in a one-out-of-two taken twice logic. Each trip system provides a closure signal to both recirculation pump discharge valves.

Low reactor water level in the shroud is detected by two additional instruments. When the level is greater than the low level setpoint, LPCI may no longer be required, therefore, other modes of RHR (e.g., suppression pool cooling) are allowed. The variable is monitored by two transmitters, which are, in turn, connected to two trip units. The outputs of the trip units are connected to relays whose contacts provide input to one of two trip systems. Each trip system provides a permissive signal to open the associated subsystems containment spray and suppression pool cooling isolation valves. Manual overrides for these isolations below the low level setpoint are provided.

Containment high pressure is detected by four instruments to automatically isolate the containment spray mode of RHR when containment depressurization is not required. This Function also precludes inadvertent diversion of LPCI flow unless containment overpressurization is indicated. This variable is monitored by four pressure switches, whose contacts

(continued)

BASES

BACKGROUND

Low Pressure Coolant Injection System (continued)

provide input to two trip systems. The outputs of the contacts are arranged in a one-out-of-two taken twice logic for each trip system. Each trip system provides an input to the associated subsystems containment spray valves.

High Pressure Coolant Injection System

The HPCI System may be initiated by either automatic or manual means, although manual initiation requires manipulation of individual pump and valve control switches. Automatic initiation occurs for conditions of Reactor Vessel Water Level – Low Low (Level 2) or Drywell Pressure – High. Each of these variables is monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each Function.

The HPCI pump discharge flow and pressure are monitored by a flow switch and pressure switch, respectively. When the pump is running (as indicated by the pressure switch) and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow assumed in the accident analysis.

The HPCI test line isolation valve is closed upon receipt of a HPCI initiation signal to allow the full system flow assumed in the accident analysis.

The HPCI System also monitors the water levels in the condensate storage tanks (CSTs) and the suppression pool because these are the two sources of water for HPCI operation. Reactor grade water in the CSTs is the normal source. The CST suction source consists of two CSTs connected in parallel to the HPCI pump suction. Upon receipt of a HPCI initiation signal, the CST suction valve is automatically signaled to open (it is normally in the open position) unless both suppression pool suction valves are open. If the water level in both CSTs falls below a

(continued)

BASES

BACKGROUND

High Pressure Coolant Injection System (continued)

preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. Two level switches are used to detect low water level in each CST. One switch associated with each CST can cause the suppression pool suction valves to open and the CST suction valve to close. The suppression pool suction valves also automatically open and the CST suction valve closes if high water level is detected in the suppression pool. Two level switches monitor suppression pool water level. Either switch can cause the suppression pool suction valves to open and the CST suction valves to close. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be full open before the other automatically closes.

The HPCI provides makeup water to the reactor until the reactor vessel water level reaches the Reactor Vessel Water Level - High (Level 8) setting, at which time the HPCI turbine trips, which causes the turbine's stop valve to close. The logic is two-out-of-two to provide high reliability of the HPCI System. The HPCI System automatically restarts if a Reactor Vessel Water Level - Low Low (Level 2) signal is subsequently received.

Automatic Depressurization System

The ADS may be initiated by either automatic or manual means, although manual initiation requires the manipulation of the hand switches associated with each ADS valve. Automatic initiation occurs when signals indicating Reactor Vessel Water Level - Low Low Low (Level 1); confirmed Reactor Vessel Water Level - Low (Level 3); and CS or LPCI Pump Discharge Pressure - High are all present and the ADS Initiation Timer has timed out. There are two transmitters for Reactor Vessel Water Level - Low Low Low (Level 1), and one transmitter for confirmed Reactor Vessel Water Level - Low (Level 3) in each of the two ADS trip systems. Each of these transmitters connects to a trip unit, which then drives a relay whose contacts form the initiation logic.

Each ADS trip system includes a time delay between satisfying the initiation logic and the actuation of the ADS

(continued)

BASES

BACKGROUND

Automatic Depressurization System (continued)

Each ADS trip system includes a time delay between satisfying the initiation logic and the actuation of the ADS valves. The ADS Initiation Timer time delay setpoint chosen is long enough that the HPCI has sufficient operating time to recover to a level above Level 1, yet not so long that the LPCI and CS Systems are unable to adequately cool the fuel if the HPCI fails to maintain that level. An alarm in the control room is annunciated when either of the timers is timing. Resetting the ADS initiation signals prior to time out of the ADS Initiation Timers resets the ADS Initiation Timers.

The ADS also monitors the discharge pressures of the four LPCI pumps and the two CS pumps. Each ADS trip system includes two discharge pressure permissive switches from one CS and from two LPCI pumps in the associated Division (i.e., Division 1 CS subsystem A and LPCI pumps A and C input to ADS trip system A, and Division 2 CS subsystem B and LPCI pumps B and D input to ADS trip system B). The signals are used as a permissive for ADS actuation, indicating that there is a source of core coolant available once the ADS has depressurized the vessel. Any one of the six low pressure pumps is sufficient to permit automatic depressurization. The switches associated with one ADS trip system also provide signals to the other ADS trip system, but these signals are not required for the other ADS trip system to be considered OPERABLE.

The ADS logic in each trip system is arranged in two strings. Each string has a contact from Reactor Vessel Water Level - Low Low Low (Level 1). One of the two strings in each trip system must also have a confirmed Reactor Vessel Water Level - Low (Level 3). All contacts in both logic strings must close, the ADS initiation timer must time out, and a CS or LPCI pump discharge pressure signal must be present to initiate an ADS trip system. Either the A or B trip system will cause all the ADS relief valves to open. Once the ADS initiation signal is present, it is individually sealed in until manually reset.

Manual inhibit switches are provided in the control room for the ADS; however, their function is not required for ADS OPERABILITY (provided ADS is not inhibited when required to be OPERABLE).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Table 3.3.5.1-1 is modified by two footnotes. Footnote (a) is added to clarify that the associated functions are required to be OPERABLE in MODES 4 and 5 only when their supported ECCS are required to be OPERABLE per LCO 3.5.2, "ECCS - Shutdown." Footnote (b) is added to show that certain ECCS instrumentation Functions also perform EDG initiation.

Allowable Values are specified for each ECCS Function specified in the table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis or other appropriate documents. The trip setpoints are derived from the analytical limits and account for all worst case instrumentation uncertainties as appropriate (e.g., drift, process effects, calibration uncertainties, and severe environmental errors (for channels that must function in harsh environments as defined by 10 CFR 50.49)). The trip setpoints derived in this manner provide adequate protection because all expected uncertainties are accounted for. The Allowable Values are then derived from the trip setpoints by accounting for normal effects that would be seen during periodic surveillance or calibration. These effects are instrumentation uncertainties observed during normal operation (e.g., drift and calibration uncertainties).

(continued)

BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY (continued)

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions that may require ECCS (or EDG) initiation to mitigate the consequences of a design basis transient or accident. To ensure reliable ECCS and EDG function, a combination of Functions is required to provide primary and secondary initiation signals.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Core Spray and Low Pressure Coolant Injection Systems

1.a, 2.a. Reactor Vessel Water Level - Low Low Low (Level 1)

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The low pressure ECCS and associated EDGs are initiated at Level 1 to ensure that core spray and flooding functions are available to prevent or minimize fuel damage. The EDGs are initiated from Function 1.a and 2.a. The Reactor Vessel Water Level - Low Low Low (Level 1) is one of the Functions assumed to be OPERABLE and capable of initiating the ECCS during the transients analyzed in Reference 3. In addition, the Reactor Vessel Water Level - Low Low Low (Level 1) Function is directly assumed in the analysis of the recirculation line break (Refs. 1, 2, and 4). The core cooling function of the ECCS, along with the scram action of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level - Low Low Low (Level 1) signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level - Low Low Low (Level 1) Allowable Value is chosen to allow time for the low pressure core flooding systems to activate and provide adequate cooling. The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the instide bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 6).

(continued)

BASES

APPLICABLE SAFETY ANALYSIS LCO, and APPLICABILITY

<u>1.a, 2.a. Reactor Vessel Water Level - Low Low Low (Level 1) (continued)</u>

bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 6).

Thus, four channels of the CS and LPCI Reactor Vessel Water Level - Low Low Low (Level 1) Function are only required to be OPERABLE when the ECCS are required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Per Footnote (a) to Table 3.3.5.1-1, this ECCS Function is only required to be OPERABLE in MODES 4 and 5 whenever the associated ECCS is required to be OPERABLE per LCO 3.5.2. Refer to LCO 3.5.1 and LCO 3.5.2, for Applicability Bases for the low pressure ECCS subsystems; LCO 3.8.1 and LCO 3.8.2, "AC Sources - Shutdown," for Applicability Bases for the EDGs.

<u>1.b, 2.b. Drywell Pressure - High</u>

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS and associated EDGs are initiated upon receipt of the Drywell Pressure - High Function in order to minimize the possibility of fuel damage. The EDGs are initiated from Function 1.b and 2.b. The Drywell Pressure - High Function, along with the Reactor Water Level - Low Low Low (Level 1) Function, is directly assumed in the analysis of the recirculation line break (Refs. 1, 2, and 4). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

The Drywell Pressure - High Function is required to be OPERABLE when the ECCS or EDG(s) are required to be OPERABLE in conjunction with times when the primary containment is required to be OPERABLE. Thus, four channels of the CS and LPCI Drywell Pressure - High Function are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude ECCS and EDG initiation. In MODES 4 and 5, the Drywell Pressure - High Function is not required, since there is insufficient energy in the reactor to pressurize the primary containment to Drywell Pressure - High setpoint. Refer to LCO 3.5.1 for

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.e. Reactor Vessel Shroud Level (Level 0) (continued)

is implicitly assumed in the analysis of the recirculation line break (Refs. 1, 2 and 4) since the analysis assumes that no LPCI flow diversion occurs when reactor water level is below Level 0.

Reactor Vessel Shroud Level (Level 0) signals are initiated from two level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. The Reactor Vessel Shroud Level (Level 0) Allowable Value is chosen to allow the low pressure core flooding systems to activate and provide adequate cooling before allowing a manual transfer. The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 6). 5

Two channels of the Reactor Vessel Shroud Level (Level 0) Function are only required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, the specified initiation time of the LPCI subsystems is not assumed, and other administrative controls are adequate to control the valves associated with this Function (since the systems that the valves are opened for are not required to be OPERABLE in MODES 4 and 5 and are normally not used).

2.h. Containment Pressure-High

The Containment Pressure-High Function is provided as an isolation of the containment spray mode of RHR on decreasing containment pressure following manual actuation of the system. This isolation ensures excessive depressurization of the containment does not occur due to containment spray actuation. This Function also serves as an interlock permissive to allow the RHR System to be manually aligned from the LPCI mode to the containment spray mode after containment pressure has exceeded the trip setting. The permissive ensures that containment pressure is elevated before the manual transfer is allowed. This ensures that LPCI is available to prevent or minimize fuel damage until such time that the operator determines that containment pressure control is needed. The Containment Pressure-High

(continued)

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.h. Containment Pressure - High (continued)

Function is implicitly assumed in the analysis of LOCAs inside containment (Refs. 1, 2, and 4) since the analysis assumes that containment spray occurs when containment pressure is high.

Containment Pressure - High signals are initiated from four pressure switches that sense drywell pressure. The Containment Pressure - High lower Allowable Value is chosen to ensure isolation of containment spray prior to a negative containment pressure occurring. This maintains margin to the negative design pressure and minimizes operation of the reactor building-to-suppression chamber vacuum breakers, which in turn prevents de-inserting the atmosphere. The upper Allowable Value is chosen to ensure containment spray is not isolated when there may be a need for containment spray.

Four channels of the Containment Pressure - High Function are only required to be OPERABLE in MODES 1, 2, and 3. In MODES 4 and 5, containment spray is not assumed to be initiated, and other administrative controls are adequate to control the valves that this Function isolates.

High Pressure Coolant Injection System

3.a. Reactor Vessel Water Level - Low Low (Level 2)

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the HPCI System is initiated at Level 2 to maintain level above the top of the active fuel. In addition, the Standby Gas Treatment (SGT) System suction valves receive an open signal so that the gland seal exhaust from the HPCI turbine can be treated. Opening of the SGT System suction valves results in automatic starting of the SGT System. The Reactor Vessel Water Level - Low Low (Level 2) is one of the Functions assumed to be OPERABLE and capable of initiating HPCI during the transients analyzed in Reference 3. Additionally, the Reactor Vessel Water Level - Low Low (Level 2) Function associated with HPCI is assumed to be OPERABLE and capable of initiating HPCI in the analysis of line breaks (Refs. 1

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

<u>3.a. Reactor Vessel Water Level - Low Low (Level 2) (continued)</u>

and 4). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level - Low Low (Level 2) signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level - Low Low (Level 2) Allowable Value is high enough such that for complete loss of feedwater flow, the Reactor Core Isolation Cooling (RCIC) System flow with HPCI assumed to fail will be sufficient to avoid initiation of low pressure ECCS at Reactor Vessel Water Level - Low Low Low (Level 1). The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 6).

The HPCI, RCIC and ATWS-RPT initiation functions (as described in Table 3.3.5.1-1, Function 3.a; Table 3.3.5.2-1, Function 1; and LCO 3.3.4.1.a including SR 3.3.4.1.4, respectively) describe the reactor vessel water level initiation function as "Low Low (Level 2)." The Allowable Values associated with the HPCI and RCIC initiation function is different from the Allowable Value associated with the ATWS-RPT initiation function as the ATWS function has a separate analog trip unit. Nevertheless, consistent with the nomenclature typically used in design documents, the "Low Low (Level 2)" is retained in describing each of these three initiation functions.

Four channels of Reactor Vessel Water Level - Low Low (Level 2) Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI initiation. Refer to LCO 3.5.1 for HPCI Applicability Bases.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.c. Reactor Vessel Water Level - High (Level 8) (continued)

Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 6).

Two channels of Reactor Vessel Water Level - High (Level 8) Function are required to be OPERABLE only when HPCI is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.d. Condensate Storage Tank Level - Low

Low level in the CSTs indicates the unavailability of an adequate supply of makeup water from this normal source. Normally the suction valve between HPCI and the CSTs is open and, upon receiving a HPCI initiation signal, water for HPCI injection would be taken from the CSTs. However, if the water level in both CSTs falls below a preselected level, the suppression pool suction valves automatically open. Opening the suppression pool suction valves causes the CST suction valve to automatically close. This ensures that an adequate supply of makeup water is available to the HPCI pump. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be full open before the CST suction valve automatically closes. The Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool.

Condensate Storage Tank Level - Low signals are initiated from four level switches (2 per CST). The logic is arranged such that one switch associated with each CST must actuate to cause the suppression pool suction valves to open and the CST suction valve to close. The Condensate Storage Tank Level - Low Function Allowable Value is high enough to ensure (15,600 gallons of water is available in each CST) adequate pump suction head while water is being taken from the CSTs.

Four channels of the Condensate Storage Tank Level - Low Function are required to be OPERABLE only when HPCI is required to be OPERABLE to ensure that no single instrument failure can preclude HPCI swap to suppression pool source. Refer to LCO 3.5.1 for HPCI Applicability Bases.

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY (continued)

Automatic Depressurization System

4.a, 5.a. Reactor Vessel Water Level-Low Low Low (Level 1)

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Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, ADS receives one of the signals necessary for initiation from this Function. The Reactor Vessel Water Level - Low Low Low (Level 1) is one of the Functions assumed to be OPERABLE and capable of initiating the ADS during the accident analyzed in References 1, 2, and 4. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level - Low Low Low (Level 1) signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Low Low (Level 1) Function are required to be OPERABLE only when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip system A, while the other two channels input to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

The Reactor Vessel Water Level - Low Low Low (Level 1) Allowable Value is chosen to allow time for the low pressure core flooding systems to initiate and provide adequate cooling. The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 6).

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4.b, 5.b. Automatic Depressurization System Initiation Timer

The purpose of the Automatic Depressurization System Initiation Timer is to delay depressurization of the reactor vessel to allow the HPCI System time to maintain reactor vessel water level. Since the rapid depressurization caused by ADS operation is one of the most severe transients on the reactor vessel, its occurrence should be limited. By delaying initiation of the ADS Function, the operator is

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.b, 5.b. Automatic Depressurization System Initiation Timer (continued)

given the chance to monitor the success or failure of the HPCI System to maintain water level, and then to decide whether or not to allow ADS to initiate, to delay initiation further by recycling the timer, or to inhibit initiation permanently. The Automatic Depressurization System Initiation Timer Function is assumed to be OPERABLE for the accident analyses of Reference 1, 2, and 4 that require ECCS initiation and assume failure of the HPCI System.

There are two Automatic Depressurization System Initiation Timer relays, one in each of the two ADS trip systems. The Allowable Value for the Automatic Depressurization System Initiation Timer is chosen so that there is still time after depressurization for the low pressure ECCS subsystems to provide adequate core cooling.

Two channels of the Automatic Depressurization System Initiation Timer Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. (One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.c, 5.c. Reactor Vessel Water Level - Low (Level 3)

The Reactor Vessel Water Level - Low (Level 3) Function is used by the ADS only as a confirmatory low water level signal. ADS receives one of the signals necessary for initiation from Reactor Vessel Water Level - Low Low Low (Level 1) signals. In order to prevent spurious initiation of the ADS due to spurious Level 1 signals, a Level 3 signal must also be received before ADS initiation commences.

Reactor Vessel Water Level - Low (Level 3) signals are initiated from two level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. The Allowable Value for Reactor Vessel Water Level - Low (Level 3) is selected to be the same as the RPS Level 3 scram Allowable Value for convenience. Refer to LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," for the Bases

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BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

4.c, 5.c. Reactor Vessel Water Level - Low (Level 3) (continued)

discussion of this Function. The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 6). I

Two channels of Reactor Vessel Water Level - Low (Level 3) Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. One channel inputs to ADS trip system A, while the other channel inputs to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

4.d, 4.e, 5.d, 5.e. Core Spray and Low Pressure Coolant Injection Pump Discharge Pressure - High

The Pump Discharge Pressure - High signals from the CS and LPCI pumps are used as permissives for ADS initiation, indicating that there is a source of low pressure cooling water available once the ADS has depressurized the vessel. Pump Discharge Pressure - High is one of the Functions assumed to be OPERABLE and capable of permitting ADS initiation during the events analyzed in References 1, 2, and 4 with an assumed HPCI failure. For these events the ADS depressurizes the reactor vessel so that the low pressure ECCS can perform the core cooling function. This core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Pump discharge pressure signals are initiated from twelve pressure switches, two on the discharge side of each of the six low pressure ECCS pumps. In order to generate an ADS permissive in one trip system, it is necessary that only one pump (both channels for the pump) indicate the high discharge pressure condition. The Pump Discharge Pressure - High Allowable Value is less than the pump discharge pressure when the pump is operating in a full flow mode and high enough to avoid any condition that results in a discharge pressure permissive when the CS and LPCI pumps are aligned for injection and the pumps are not running.

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BASES

APPLICABLE SAFETY ANALYSIS LCO, and APPLICABILITY

4.d, 4.e, 5.d, 5.e. Core Spray and Low Pressure Coolant Injection Pump Discharge Pressure – High (continued)

The actual operating point of this function is not assumed in any transient or accident analysis. However, this function is implicitly assumed to operate to provide the ADS permissive to depressurize the RCS to allow the ECCS low pressure systems to operate.

Twelve channels of Core Spray and Low Pressure Coolant Injection Pump Discharge Pressure–High Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two CS channels associated with CS pump A and four LPCI channels associated with LPCI pumps A and C are required for trip system A. Two CS channels associated with CS pump B and four LPCI channels associated with LPCI pumps B and D are required for trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

ACTIONS

A Note has been provided to modify the ACTIONS related to ECCS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable ECCS instrumentation channels provide appropriate compensatory measures for separate inoperable Condition entry for each inoperable ECCS instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.5.1-1. The applicable Condition referenced in the table is Function dependent. Each time a channel is discovered inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

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BASES

ACTIONS

B.1, B.2, and B.3 (continued)

Notes are also provided (Note 2 to Required Action B.1 and the Note to Required Action B.2) to delineate which Required Action is applicable for each Function that requires entry into Condition B if an associated channel is inoperable. This ensures that the proper loss of initiation capability check is performed. Required Action B.1 (the Required Action for certain inoperable channels in the low pressure ECCS subsystems) is not applicable to Functions 2.e and 2.h, since these Functions provide backup to administrative controls ensuring that operators do not divert LPCI flow from injecting into the core when needed, and do not spray the containment unless needed. Thus, a total loss of Function 2.e or 2.h capability for 24 hours is allowed, since the LPCI subsystems remain capable of performing their intended function. 1J

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action B.1, the Completion Time only begins upon discovery that a redundant feature in the same system (e.g., both CS subsystems) cannot be automatically initiated due to inoperable, untripped channels within the same Function as described in the paragraph above. For Required Action B.2, the Completion Time only begins upon discovery that the HPCI System cannot be automatically initiated due to two inoperable, untripped channels for the associated Function in the same trip system. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 7) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.3. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue.

(continued)

BASES

ACTIONS

F.1 and F.2 (continued)

result in redundant automatic initiation capability being lost for the ADS. Redundant automatic initiation capability is lost if either (a) one Function 4.a channel and one Function 5.a channel are inoperable and untripped, or (b) one Function 4.c channel and one Function 5.c channel are inoperable and untripped.

In this situation (loss of automatic initiation capability), the 96 hour or 8 day allowance, as applicable, of Required Action F.2 is not appropriate and all ADS valves must be declared inoperable within 1 hour after discovery of loss of ADS initiation capability.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action F.1, the Completion Time only begins upon discovery that the ADS cannot be automatically initiated due to inoperable, untripped channels within similar ADS trip system Functions as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 8 days has been shown to be acceptable (Ref. 7) to permit restoration of any inoperable channel to OPERABLE status if both HPCI and RCIC are OPERABLE. If either HPCI or RCIC is inoperable, the time is shortened to 96 hours. If the status of HPCI or RCIC changes such that the Completion Time changes from 8 days to 96 hours, the 96 hours begins upon discovery of HPCI or RCIC inoperability. However, the total time for an inoperable, untripped channel cannot exceed 8 days. If the status of HPCI or RCIC changes such that the Completion Time changes from 96 hours to 8 days, the "time zero" for beginning the 8 day "clock" begins upon discovery of the inoperable, untripped channel. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK guarantees that undetected outright channel failure is limited to 12 hours; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Channel agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. 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 3.3.5.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1.2 (continued)

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analyses of Reference 7.

SR 3.3.5.1.3 and SR 3.3.5.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of SR 3.3.5.1.3 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 3.3.5.1.5 is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. | J

SR 3.3.5.1.4

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.5.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analyses. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than the setting accounted for in the appropriate setpoint methodology.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1.4 (continued)

The Frequency of 184 days is based on the reliability, accuracy, and lower failure rates of the associated solid-state electronic Analog Transmitter/Trip System components.

SR 3.3.5.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.1, LCO 3.5.2, LCO 3.8.1, and LCO 3.8.2 overlaps this Surveillance to complete testing of the assumed safety function.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. UFSAR, Section 6.5.
2. UFSAR, Section 14.6.
3. UFSAR, Section 14.5.
4. NEDC-31317P, Revision 2, James A. FitzPatrick Nuclear Power Plant, SAFER/GESTR-LOCA, Loss-of-Coolant Accident Analysis, April 1993.
5. 10 CFR 50.36(c)(2)(ii).
6. Drawing 11825-5.01-15D, Rev. D, Reactor Assembly Nuclear Boiler, (GE Drawing 919D690BD).
7. NEDC-30936P-A, BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation), Part 2, December 1988.

DB2 INSERT ASA

A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis or other appropriate documents. The trip setpoints are derived from the analytical limits and account for all worst case instrumentation uncertainties as appropriate (e.g., drift, process effects, calibration uncertainties, and severe environmental errors (for channels that must function in harsh environments as defined by 10 CFR 50.49)). The trip setpoints derived in this manner provide adequate protection because all expected uncertainties are accounted for. The Allowable Values are then derived from the trip setpoints by accounting for normal effects that would be seen during periodic surveillance or calibration. These effects are instrumentation uncertainties observed during normal operation (e.g., drift and calibration uncertainties). PA2

DB5 INSERT Function 1

The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 2). 5

The HPCI, RCIC and ATWS-RPT initiation functions (as described in Table 3.3.5.1-1, Function 3.a; Table 3.3.5.2-1, Function 1; and LCO 3.3.4.1.a including SR 3.3.4.1.4, respectively) describe the reactor vessel water level initiation function as "Low Low (Level 2)." The Allowable Values associated with the HPCI and RCIC initiation function is different from the Allowable Value associated with the ATWS-RPT initiation function as the ATWS function has a separate analog trip unit. Nevertheless, consistent with the nomenclature typically used in design documents, the "Low Low (Level 2)" is retained in describing each of these three initiation functions. 3

DB1

INSERT Function 2

The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 2). J

BASES

ACTIONS

B.1 and B.2 (continued)

inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition E must be entered and its Required Action taken.

C.1

3 1 XI

A risk based analysis was performed and determined that an allowable out of service time of 24 hours (Ref. 1) is acceptable to permit restoration of any inoperable channel to OPERABLE status (Required Action C.1). A Required Action (similar to Required Action B.1) limiting the allowable out of service time, if a loss of automatic RCIC initiation capability exists, is not required. This Condition applies to the Reactor Vessel Water Level—High, Level 8 Function whose logic is arranged such that any inoperable channel will result in a loss of automatic RCIC initiation capability due to closure of the RCIC steam inlet valve. As stated above, this loss of automatic RCIC initiation capability was analyzed and determined to be acceptable. This Condition also applies to the Manual Initiation Function. Since this Function is not assumed in any accident or transient analysis, a total loss of manual initiation capability (Required Action C.1) for 24 hours is allowed. The Required Action does not allow placing a channel in trip since this action would not necessarily result in a safe state for the channel in all events.

PA3 DB1

D.1, D.2.1, and D.2.2

associated with the same CST

DB1

Required Action D.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in automatic component initiation capability being lost for the feature(s). For Required Action D.1, the RCIC System is the only associated feature. In this case, automatic initiation capability (automatic suction source alignment) is lost if two Function 3 channels ~~or two Function 4 channels~~ are inoperable and untripped. In this situation (loss of automatic suction ~~swap~~ source alignment), the 24 hour allowance of Required Actions D.2.1 and D.2.2 is not

DB3 DB3 PA1

(continued)

Revision 3

TA1 INSERT SR 3.3.5.2.2

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. | J

BASES (continued) GENE (–) PA5 J

REFERENCES X1 numbering DB1

③ ~~1~~. ~~NEDE~~-770-06-2~~,~~ Ⓐ "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," ~~February 1991~~. December 1992 DB5 J

1. 10 CFR 50.36 (c)(2)(ii). X1
2. Drawing 11825-5.01-15D, Rev. D, Reactor Assembly Nuclear Boiler, (GE Drawing 919D690BD). DB1

J

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Reactor Vessel Water Level - Low Low (Level 2) (continued)

water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level - Low Low (Level 2) Allowable Value is set high enough such that for complete loss of feedwater flow, the RCIC System flow with high pressure coolant injection assumed to fail will be sufficient to avoid initiation of low pressure ECCS at Level 1. The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 2). J

The HPCI, RCIC and ATWS-RPT initiation functions (as described in Table 3.3.5.1-1, Function 3.a; Table 3.3.5.2-1, Function 1; and LCO 3.3.4.1.a including SR 3.3.4.1.4, respectively) describe the reactor vessel water level initiation function as "Low Low (Level 2)." The Allowable Values associated with the HPCI and RCIC initiation function is different from the Allowable Value associated with the ATWS-RPT initiation function as the ATWS function has a separate analog trip unit. Nevertheless, consistent with the nomenclature typically used in design documents, the "Low Low (Level 2)" is retained in describing each of these three initiation functions. J

Four channels of Reactor Vessel Water Level - Low Low (Level 2) Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC initiation. Refer to LCO 3.5.3 for RCIC Applicability Bases.

2. Reactor Vessel Water Level - High (Level 8)

High RPV water level indicates that sufficient cooling water inventory exists in the reactor vessel such that there is no danger to the fuel. Therefore, the Level 8 signal is used to close the RCIC steam inlet valve to prevent overflow into the main steam lines (MSLs).

Reactor Vessel Water Level - High (Level 8) signals for RCIC are initiated from two level transmitters from the narrow range water level measurement instrumentation, which sense

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. Reactor Vessel Water Level - High (Level 8) (continued)

the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Both Level 8 signals are required in order to close the RCIC steam inlet valve.

The Reactor Vessel Water Level - High (Level 8) Allowable Value is high enough to preclude isolating the steam inlet valve during normal operation, yet low enough to prevent water overflowing into the MSLs. The Allowable Value is referenced from a level of water 352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 2). 5

Two channels of Reactor Vessel Water Level - High (Level 8) Function are available and are required to be OPERABLE when RCIC is required to be OPERABLE to ensure that no single instrument failure can preclude RCIC initiation. Refer to LCO 3.5.3 for RCIC Applicability Bases.

3. Condensate Storage Tank (CST) Level - Low

Low level in the CSTs indicates the unavailability of an adequate supply of makeup water from this normal source. Normally, the suction valve between the RCIC pump and the CSTs is open and, upon receiving a RCIC initiation signal, water for RCIC injection would be taken from the CSTs. However, if the water level in both CSTs falls below a preselected level, first the suppression pool suction valves automatically open, and then the CSTs suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the RCIC pump. To prevent losing suction to the pump, the suction valves are interlocked so that the suppression pool suction valves must be open before the CSTs suction valve automatically closes.

Two level switches are used to detect low water level in each CST. The Condensate Storage Tank Level - Low Function Allowable Value is set high enough (15,600 gallons of water is available in each CST) to ensure adequate pump suction head while water is being taken from the CST.

Four channels of Condensate Storage Tank Level - Low Function are available and are required to be OPERABLE when RCIC is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.2.1 (continued)

channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. 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 3.3.5.2.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. |J

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 3.

SR 3.3.5.2.3 and SR 3.3.5.2.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of SR 3.3.5.2.3 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

REFERENCES

1. 10 CFR 50.36(c)(2)(ii).
2. Drawing 11825-5.01-15D, Rev. D, Reactor Assembly Nuclear Boiler, (GE Drawing 919D690BD).
3. GENE-770-06-2-A, Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications, December 1992.

TECHNICAL CHANGES - MORE RESTRICTIVE

M12 (continued)

condition during the performance of a required Surveillance. This change is more restrictive on plant operation but necessary to ensure the PCIVs will isolate the penetration flow path when necessary, consistent with the analyses.

M13 The CTS Applicability of the Primary Containment Isolation Functions as described in CTS 3.2.A is whenever primary containment integrity is required. The Applicability identified in CTS Table 3.2-1 Note 1 is whenever Primary Containment integrity is required by Specification 3.7.A.2. The Applicability in CTS 3.7.A.2 is whenever the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel. In addition, there is an exception in CTS 3.7.A.2, to not require primary containment integrity to be met during low power physics tests at atmospheric pressure and power levels not to exceed 5 MWt, however any change to this requirement is discussed in the Discussion of Changes for ITS 3.10.8. The scope of the current Applicability covers MODE 1, 3 and portions of MODE 2 operations. In general the Applicability of most Functions in the ITS will be MODES 1, 2 and 3. This change is considered more restrictive since the Functions will be required to be Operable at all times in MODE 2 (which is consistent with current practice). Changes to the current Applicability are further discussed in L3, M2, and L17. This change is consistent with NUREG-1433, Revision 1.

M14 This change replaces the following setpoints or Allowable Values (A16) in the CTS:

(1) HPCI Steam Line Low Pressure Isolation in CTS Table 3.2-1, Item 14, of $100 > P > 50$ psig to ≥ 61 psig and ≤ 90 psig (Function 3.b for HPCI);

(2) Main Steam Line Leak Detection High Temperature Isolation in CTS Table 3.2-1, Item 10, of < 40°F above max. ambient to ≤ 195°F (Function 1.e);

(3) HPCI and RCIC Steam Line/Area Temperature Isolation in CTS Table 3.2-1, Item 16 (HPCI) and Item 20 (RCIC), from ≤ 40°F above max. ambient to:

(a) ≤ 160°F (Function 3.d) for HPCI Steam Line Penetration (Drywell Entrance) Area Temperature – High,

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA12 (continued)

These operational details are not necessary to ensure the PCI instrumentation is OPERABLE. The requirements of ITS 3.3.6.1, which require the PCI instrumentation to be OPERABLE, and the definition of OPERABILITY suffice. As such, these details are not required to be in the ITS to provide adequate protection of public health and safety. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 The CTS Safety Limit and actions in CTS 1/2.2.2, when operating the RHR System in the Shutdown Cooling Mode, are proposed to be incorporated into ITS 3.3.6.1 (Table 3.3.6.1-1 for Primary Containment Isolation Instrumentation). The RHR Shutdown Cooling System is designed with an interlock in the logic for the system isolation valves, which are normally closed during power operation, to prevent opening of the valves above a preset pressure setpoint (Allowable Value). This setpoint is selected to assure that pressure integrity of the RHR system is maintained. The CTS 1.2.2 requirement that the pressure be less than the limit "when operating the Residual Heat Removal Pump" is covered by the Applicability of the instrumentation, which is MODES 1, 2, and 3 (when primary containment is required Operable). In MODES 4 and 5 with the pump operating, the reactor is depressurized and the potential for inadvertent pressurization is very low. Additionally, the context of CTS 2.2.2 is covered by proposed ACTION F which requires that the affected penetration flow path(s) be isolated. The high pressure interlock is only provided for equipment protection to prevent an inter-system LOCA and, as such, this function should not be considered a Safety Limit on plant operation. | J

L2 The details relating to the Instrument I.D. numbers for the containment isolation instrumentation in CTS 4.2.A and CTS Table 3.2-8 for Function 4 (Containment High Range Radiation Monitor) are proposed to be deleted. These details are not necessary to ensure the containment isolation instrumentation is maintained Operable. The requirements of ITS 3.3.6.1 (which describes the instrumentation) and the associated Surveillance Requirements are adequate to ensure the required instrumentation is maintained Operable. The Bases also provide a description of the type of instrumentation required by the Specification.

L3 The Applicability for the CTS Table 3.2-1 Reactor Low Water Level Function is MODES 1, 2, and 3 (when primary containment is required) as