BASES

APPLICABLE SAFETY ANALYSIS (continued)

margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 14 of the UFSAR.

A plant specific LOCA analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Ref. 3).

The transient analyses of Chapter 14 of the UFSAR have also been performed for single recirculation loop operation (Ref. 3) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) and the control rod block instrumentation Allowable Values are also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR limits for single loop operation are specified in the COLR. The APRM Neutron Flux - High (Flow Biased) Allowable Value is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation." The Rod Block Monitor - Upscale Allowable Value is specified in LCO 3.3.2.1, "Control Rod Block Instrumentation."

Operation of the Reactor Water Recirculation System also ensures adequate core flow at higher power levels such that conditions conducive to the onset of thermal hydraulic instability are avoided. The UFSAR Section 16.6 (Ref. 4) requires protection of fuel thermal safety limits from conditions caused by thermal hydraulic instability. Thermal hydraulic instabilities can result in power oscillations which could result in exceeding the MCPR Safety Limit. The MCPR Safety Limit is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). Implementation of operability requirements for avoidance of, and protection from thermal-hydraulic instability, consistent with the BWR Owners' Group Long-Term Stability Solution Option I-D (Refs. 5 and 6) provides assurance that power oscillations are either prevented or can be readily detected and suppressed without exceeding the specified acceptable fuel design limits. To

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

minimize the likelihood of thermal-hydraulic instability which results in power oscillations, a power-to-flow "Exclusion Region" is calculated using the approved methodology specified in Specification 5.6.5. The resulting "Exclusion Region" may change each fuel cycle and is therefore specified in the COLR. Entries into the "Exclusion Region" may occur as a result of an abnormal event, such as a single recirculation pump trip, loss of feedwater heating, or be required to prevent equipment damage.

The core-wide mode of oscillation in the neutron flux is more readily detected (and suppressed) than the regional mode of oscillation due to the spatial averaging of the Average Power Range Monitor (APRM). The Option I-D analysis for JAFNPP (Ref. 7) demonstrates that this protection is provided at a high statistical confidence level for regional mode oscillations. Reference 7 also demonstrates that the core-wide mode of oscillation is more likely to occur rather than regional oscillations due to the large single-phase pressure drop associated with the small fuel inlet orifice diameters.

Recirculation loops operating satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 8).

LCO

Two recirculation loops are required to be in operation with their flows matched within the limits specified in SR 3.4.1.2 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. Alternatively, with only one recirculation loop in operation, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), APRM Neutron Flux–High (Flow Biased) Allowable Value (LCO 3.3.1.1) and the Rod Block Monitor–Upscale Allowable Value (LCO 3.3.2.1) must be applied to allow continued operation consistent with the assumptions of Reference 3. In addition, during two-loop and single-loop operation, the combination of core flow and THERMAL POWER must be outside the Exclusion Region of

(continued)

BASES

LCO (continued)

the power-to-flow map specified in the COLR to ensure core thermal-hydraulic instability does not occur.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Water Recirculation System are necessary since there is considerable energy in the reactor core, core thermal-hydraulic instability may occur, and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

A.1

With the reactor operating at core flow and THERMAL POWER conditions within the Exclusion Region of the power-to-flow map it is in a condition where thermal-hydraulic instabilities are conservatively predicted to occur, and must be brought to an operating state where such instabilities are not predicted to occur. To achieve this status, action must be taken immediately to exit the Exclusion Region. This is accomplished by inserting control rods or increasing core flow such that the combination of THERMAL POWER and core flow move to a point outside the Exclusion Region. The action is considered sufficient to preclude core thermal-hydraulic instabilities which could challenge the MCPR safety limit. The starting of a recirculation pump is not used as a means to exit the Exclusion Region of the power-to-flow map. Starting an idle recirculation pump could result in a reduction in inlet core enthalpy and enhance conditions necessary for thermal-hydraulic instabilities.

B.1

With the requirements of the LCO not met for reasons other than Condition A, the recirculation loops must be restored to operation with matched flows within 24 hours. A recirculation loop is considered not in operation when the

(continued)

BASES

ACTIONS

C.1 (continued)

the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR ensures the combination of core flow and THERMAL POWER are within appropriate limits to prevent uncontrolled thermal-hydraulic oscillations. At low recirculation flows and high reactor power, the reactor exhibits increased susceptibility to thermal-hydraulic instability. The power-to-flow map specified in the COLR is based on guidance provided in Reference 7. The 12 hour Frequency is based on operating experience and the operator's knowledge of the reactor status, including significant changes in THERMAL POWER and core flow. J

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the APRM Neutron Flux—High (Startup) Function in LCO 3.3.1.1 will prevent operation in the Exclusion Region while in MODE 2. J

SR 3.4.1.2

This SR ensures the recirculation loops are within the allowable limits for mismatch. At low core flow (i.e., < 70% of rated core flow), the MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.2 (continued)

Condition B must be entered, and the loop with the lower flow must be declared "not in operation". (However, for the purpose of performing SR 3.4.1.1, the flow rate of both loops shall be used.) The SR is not required when only one loop is in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

REFERENCES

1. UFSAR, Section 14.6.

2. UFSAR, Section 14.5.

3. NEDO-24281, FitzPatrick Nuclear Power Plant Single-Loop Operation, August 1980.

4. UFSAR, Section 16.6.

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, March 1992.

7. GENE-637-044-0295, Application Of The "Regional Exclusion With Flow-Biased APRM Neutron Flux Scram" Stability Solution (Option I-D) To The James A. FitzPatrick Nuclear Power Plant, February 1995.

8. 10 CFR 50.36(c)(2)(ii).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The capability of reflooding the core to two-thirds core height is dependent upon the structural integrity of the jet pumps. If the structural system, including the beam holding a jet pump in place, fails, jet pump displacement and performance degradation could occur, resulting in an increased flow area through the jet pump and a lower core flooding elevation. This could adversely affect the water level in the core during the reflood phase of a LOCA as well as the assumed blowdown flow during a LOCA.

Jet pumps satisfy Criterion 2 of the NRC Policy Statement. 10 CFR 50.36 (c)(2)(ii) (Ref. 2) X1 J

LCO

The structural failure of any of the jet pumps could cause significant degradation in the ability of the jet pumps to allow reflooding to two-thirds core height during a LOCA. OPERABILITY of all jet pumps is required to ensure that operation of the Reactor Coolant Water Recirculation System will be consistent with the assumptions used in the licensing basis analysis (Ref. 1). PA1

APPLICABILITY

In MODES 1 and 2, the jet pumps are required to be OPERABLE since there is a large amount of energy in the reactor core and since the limiting DBAs are assumed to occur in these MODES. This is consistent with the requirements for operation of the Reactor Coolant Water Recirculation System (LCO 3.4.1). PA1 J

In MODES 3, 4, and 5, the Reactor Coolant Recirculation System is not required to be in operation, and when not in operation, sufficient flow is not available to evaluate jet pump OPERABILITY.

ACTIONS

A.1

An inoperable jet pump to can increase the blowdown area and reduce the capability of reflooding during a design basis LOCA. If one or more of the jet pumps are inoperable, 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 MODE 3 within 12 hours. The Completion Time of 12 hours is PA2

(continued)

Revision J

PA1 DB1

BASES (continued)

REFERENCES

1. FSAR, Section ~~6.3~~ 14.6.

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

~~2.~~ 3. GE Service Information Letter No. 330, including Supplement 1, Jet Pump Beam Cracks, June 9, ~~1990~~ 1980.

~~3.~~ 4. NUREG/CR-3052, Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure, November 1984.

X1 PA2 PA2 PA2 J J

Revision J

BASES

APPLICABLE SAFETY ANALYSES (continued)

The capability of reflooding the core to two-thirds core height is dependent upon the structural integrity of the jet pumps. If the structural system, including the beam holding a jet pump in place, fails, jet pump displacement and performance degradation could occur, resulting in an increased flow area through the jet pump and a lower core flooding elevation. This could adversely affect the water level in the core during the reflood phase of a LOCA as well as the assumed blowdown flow during a LOCA.

Jet pumps satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

LCO

The structural failure of any of the jet pumps could cause significant degradation in the ability of the jet pumps to allow reflooding to two-thirds core height during a LOCA. OPERABILITY of all jet pumps is required to ensure that operation of the Reactor Water Recirculation System will be consistent with the assumptions used in the licensing basis analysis (Ref. 1).

APPLICABILITY

In MODES 1 and 2, the jet pumps are required to be OPERABLE since there is a large amount of energy in the reactor core and since the limiting DBAs are assumed to occur in these MODES. This is consistent with the requirements for operation of the Reactor Water Recirculation System (LCO 3.4.1).

In MODES 3, 4, and 5, the Reactor Water Recirculation System is not required to be in operation, and when not in operation, sufficient flow is not available to evaluate jet pump OPERABILITY.

ACTIONS

A.1

An inoperable jet pump can increase the blowdown area and reduce the capability to reflood during a design basis LOCA. If one or more of the jet pumps are inoperable, the plant must be brought to a MODE in which the LCO does not apply.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1 (continued)

The deviations from normal are considered indicative of a potential problem in the recirculation drive flow or jet pump system (Ref. 3). Normal flow ranges and established jet pump differential pressure patterns are established by plotting historical data as discussed in Reference 3.

The 24 hour Frequency has been shown by operating experience to be timely for detecting jet pump degradation and is consistent with the Surveillance Frequency for recirculation loop OPERABILITY verification.

This SR is modified by two Notes. Note 1 allows this Surveillance not to be performed until 4 hours after the associated recirculation loop is in operation, since these checks can only be performed during jet pump operation. The 4 hours is an acceptable time to establish conditions appropriate for data collection and evaluation.

Note 2 allows this SR not to be performed until 24 hours after THERMAL POWER exceeds 25% of RTP. During low flow conditions, jet pump noise approaches the threshold response of the associated flow instrumentation and precludes the collection of repeatable and meaningful data. The 24 hours is an acceptable time to establish conditions appropriate to perform this SR.

REFERENCES

1. UFSAR, Section 14.6.
2. 10 CFR 50.36(c)(2)(ii).
3. GE Service Information Letter No. 330, including Supplement 1, Jet Pump Beam Cracks, June 9, 1980. J
4. NUREG/CR-3052, Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure, November 1984.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Safety/Relief Valves (S/RVs)

BASES

(Ref. 1) DB1

BACKGROUND

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of S/RVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

PA1

However for the purposes of this LCO, only the safety mode is required,

The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. ¶ The S/RVs can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (or spring mode of operation), the spring loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. This satisfies the Code requirement.

PA1

Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool. ~~The S/RVs that provide the relief mode are the low-low set (LLS) valves and the Automatic Depressurization System (ADS) valves. The LLS requirements are specified in LCO 3.6.1.6, "Low-Low Set (LLS) Valves," and the ADS requirements are specified in LCO 3.5.1, "ECCS—Operating."~~ DB2

INSERT BK6D

DB2

APPLICABLE SAFETY ANALYSES

CLB1

The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient (5) is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. ~~1~~ 3 and 4). For the purpose of the analyses, ~~[11]~~ nine S/RVs are assumed to operate in the safety mode. The analysis results demonstrate that (Ref. 4) ~~the design S/RV capacity is~~ nine S/RVs are capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design DB1 J

DB1 DB2 J

(continued)

DB2 Insert BKGD

Each S/RV can be opened manually in the relief mode from the control room by its associated two-position switch. If one of these switches is placed in the open position the logic output will energize the associated S/RV solenoid control valve directing the pneumatic supply to open the valve. Seven of these S/RV solenoid control valves can also be energized by the relay logic associated with the Automatic Depressurization System (ADS). ADS requirements are specified in LCO 3.5.1, "ECCS - Operating." In addition each S/RV can be manually operated from another control switch located at the ADS auxiliary panel located outside the control room. These switches will energize a different S/RV solenoid control valve. The details of S/RVs pneumatic supply and mechanical operation in the relief mode are described in Reference 2. 1J

BASES

(at the vessel bottom) PA1

APPLICABLE SAFETY ANALYSES (continued)

pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the ~~Design Basis Event~~ most severe pressurization transient. PA1

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. Reference ~~2~~ 5 discusses additional events that are expected to actuate the S/RVs. DB1

S/RVs satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (Ref. 6). X1

CLB1

LCO

The safety function of ~~11~~ Nine S/RVs are required to be OPERABLE to satisfy the assumptions of the safety analysis (Refs. ~~1~~ 3 and ~~2~~ 4). The requirements of this LCO are applicable only to the capability of the S/RVs to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety function). DB1

The single nominal S/RV setpoint ~~s are~~ is established (Ref. 2) to ensure that the ASME Code limit on peak reactor pressure is satisfied. ~~The ASME Code specifications require the lowest safety valve setpoint to be at or below vessel design pressure (1250 psig) and the highest safety valve to be set so that the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions.~~ The single nominal S/RV setpoint is set below the RPV design pressure (1250 psig) in accordance with ASME Code requirements. DB2 DB1 The transient evaluations in ~~the FSAR~~ Reference 5 are based on ~~these~~ this single setpoint~~s~~, but also include the additional uncertainties of ± ~~1~~ 3% of the nominal setpoint ~~drift~~ to provide an added degree of conservatism. DB2

Operation with fewer valves OPERABLE than specified, or with setpoints outside the ~~ASME~~ analysis limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded. PA1

CLB1

APPLICABILITY

In MODES 1, 2, and 3, ~~all~~ nine S/RVs must be OPERABLE, since considerable energy may be in the reactor core and the limiting design basis transients are assumed to occur in these MODES. The S/RVs may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the core heat.

(continued)

Revision 5

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

PA2 PA1

This Surveillance requires that the [required] S/RVs will open at the pressures assumed in the safety analysis of Reference 4. The demonstration of the S/RV ~~safe~~ lift safety function J settings must be performed during shutdown, since this is a bench test, to be done in accordance with the Inservice Testing Program. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint is ± [3]% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

s 3 and 4 DB1 X2 PA5

~~The 18 month Frequency was selected because this Surveillance must be performed during shutdown conditions and is based on the time between refuelings.~~ X2

CLB3

SR 3.4.3.2

PA2

A manual actuation of each [required] S/RV is performed while bypassing main steam flow to the condenser and observing ≥ 10% closure of the turbine bypass valves to verify that, mechanically, the valve is functioning properly and no blockage exists in the valve discharge line. This can also be demonstrated by the response of the turbine control valves ~~or bypass valves~~, by a change in the measured steam flow, or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the S/RVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is ~~[920]~~ 970 psig (the pressure consistent with vendor recommendations ~~recommended by the valve manufacturer~~). Adequate steam flow is represented by ~~[at least 1.25]~~ two or more turbine bypass valves open, or total steam flow ≥ 10^6 lb/hr. These conditions will require the plant to be in MODE 1, which has been shown to be an acceptable condition to perform this test. This test causes a small neutron flux transient which may cause a scram in MODE 2 (Startup) while operating close to the Average Power Range Monitors Neutron Flux - High (Startup) Allowable Value. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME Code requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure and flow are ~~is~~ reached is sufficient to achieve stable steam

CLB3 DB3 DB3 DB3 PA1

(continued)

Revision J

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.2 (continued)

conditions for testing and provides a reasonable time to complete the SR. If a valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the S/RV is considered OPERABLE.

24

The [18] 24 month on a STAGGERED TEST BASIS Frequency ensures that each solenoid for each S/RV is alternately tested. The 18 24 month Frequency was developed based on the S/RV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 3 7). Operating experience has shown that these components usually pass the Surveillance when performed at the 18 24 month Frequency. Therefore, the Frequency ~~was concluded to be~~ is acceptable from a reliability standpoint. X3 J PA1 DB1

REFERENCES

2 1. U FSAR, Section [5.2.2.2.4] 4.4. DB1 PA3

DB1 2. U FSAR, Section [15] 14.5.1.2.

7 3 3. ASME, Boiler and Pressure Vessel Code, Section XI. J

1. ASME, Boiler and Pressure Vessel Code, Section III. DB1 J

4. UFSAR, Section 16.9.3.2.3. DB1

5. UFSAR, Section 14.5.2. DB1 J

6. 10 CFR 50.36(c)(2)(ii). X1 J

Revision J

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Safety/Relief Valves (S/RVs)

BASES

BACKGROUND

The ASME Boiler and Pressure Vessel Code (Ref. 1) requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of S/RVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool.

The S/RVs can actuate by either of two modes: the safety mode or the relief mode. However, for the purposes of this LCO, only the safety mode is required. In the safety mode (or spring mode of operation), the spring loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. This satisfies the Code requirement.

Each S/RV can be opened manually in the relief mode from the control room by its associated two-position switch. If one of these switches is placed in the open position the logic output will energize the associated S/RV solenoid control valve directing the pneumatic supply to open the valve. Seven of these S/RV solenoid control valves can also be energized by the relay logic associated with the Automatic Depressurization System (ADS). ADS requirements are specified in LCO 3.5.1, "ECCS—Operating." In addition each S/RV can be manually operated from another control switch located at the ADS auxiliary panel located outside the control room. These switches will energize a different S/RV solenoid control valve. The details of S/RVs pneumatic supply and mechanical operation in the relief mode are described in Reference 2.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Refs. 3 and 4). For the purpose of the analyses (Ref. 4), nine S/RVs are assumed to operate in the safety mode. The analysis results demonstrate that nine S/RVs are capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig (at the vessel bottom) is met during the most severe pressurization transient.

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. Reference 5 discusses additional events that are expected to actuate the S/RVs.

S/RVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 6).

LCO

The safety function of nine S/RVs are required to be OPERABLE to satisfy the assumptions of the safety analysis (Refs. 3 and 4). The requirements of this LCO are applicable only to the capability of the S/RVs to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety function).

The single nominal S/RV setpoint is established (Ref. 2) to ensure that the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve setpoint to be at or below vessel design pressure (1250 psig) and the highest safety valve to be set so that the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The single nominal S/RV setpoint is set below the RPV design pressure (1250 psig) in accordance with ASME Code requirements. The transient evaluations in Reference 5 are based on this single setpoint, but also include the additional uncertainties of ± 3% of the nominal setpoint to provide an added degree of conservatism.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This Surveillance requires that the required S/RVs open at the pressures assumed in the safety analysis of References 3 and 4. The demonstration of the S/RV safety function lift settings must be performed during shutdown, since this is a bench test, to be done in accordance with the Inservice Testing Program. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint is ± 3% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

SR 3.4.3.2

A manual actuation of each required S/RV is performed while bypassing main steam flow to the condenser and observing ≥ 10% closure of the turbine bypass valves to verify that, mechanically, the valve is functioning properly and no blockage exists in the valve discharge line. This can also be demonstrated by the response of the turbine control valves, by a change in the measured steam flow, or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the S/RVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is 970 psig (the pressure consistent with vendor recommendations). Adequate steam flow is represented by two or more turbine bypass valves open, or total steam flow ≥ 10^6 lb/hr. These conditions will require the plant to be in MODE 1, which has been shown to be an acceptable condition to perform this test. This test causes a small neutron flux transient which may cause a scram in MODE 2 while operating close to the Average Power Range Monitors Neutron Flux-High (Startup) Allowable Value. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME Code requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.2 (continued)

flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required steam pressure and flow are reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If a valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the S/RV is considered OPERABLE.

The 24 month on a STAGGERED TEST BASIS Frequency ensures that each solenoid for each S/RV is alternately tested. The 24 month Frequency was developed based on the S/RV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 7). Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. UFSAR, Section 4.4.
3. UFSAR, Section 14.5.1.2.
4. UFSAR, Section 16.9.3.2.3.
5. UFSAR, Section 14.5.2.
6. 10 CFR 50.36(c)(2)(ii).
7. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Operational LEAKAGE

BASES

BACKGROUND

The RCS includes systems and components that contain or transport the coolant to or from the reactor core. The pressure containing components of the RCS and the portions of connecting systems out to and including the isolation valves define the reactor coolant pressure boundary (RCPB). The joints of the RCPB components are welded or bolted. Some joints in ≤1 inch piping are also threaded. DB3

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. Limits on RCS operational LEAKAGE are required to ensure appropriate action is taken before the integrity of the RCPB is impaired. This LCO specifies the types and limits of LEAKAGE. This protects the RCS pressure boundary described in 10 CFR 50.2, 10 CFR 50.55a(c), and ~~GDC 55 of 10 CFR 50, Appendix A~~ UFSAR, Section 16.6 (Refs. 1, 2, and 3). J DB1 PA2

The safety significance of RCS LEAKAGE from the RCPB varies widely depending on the source, rate, and duration. Therefore, detection of LEAKAGE in the ~~primary containment~~ drywell is necessary. Methods for quickly separating the identified LEAKAGE from the unidentified LEAKAGE are necessary to provide the operators quantitative information to permit them to take corrective action should a leak occur that is detrimental to the safety of the facility or the public. PA1

A limited amount of leakage inside ~~primary containment~~ the drywell is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected and isolated from the primary containment atmosphere, if possible, so as not to mask RCS operational LEAKAGE detection. J

This LCO deals with protection of the RCPB from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident.

(continued)

Typ. All Pages

Revision J

BASES (continued)

APPLICABLE SAFETY ANALYSES

The allowable RCS operational LEAKAGE limits are based on the predicted and experimentally observed behavior of pipe cracks. The normally expected background LEAKAGE due to equipment design and the detection capability of the instrumentation for determining system LEAKAGE were also considered. The evidence from experiments suggests that, for LEAKAGE even greater than the specified unidentified LEAKAGE limits, the probability is small that the imperfection or crack associated with such LEAKAGE would grow rapidly.

The unidentified LEAKAGE flow limit allows time for corrective action before the RCPB could be significantly compromised. The 5 gpm limit is a small fraction of the calculated flow from a critical crack in the primary system piping. Crack behavior from experimental programs (Refs. 2 and 3) shows that leakage rates ~~of hundreds of gallons per minute~~ much greater than 5 gpm will precede crack instability (Ref. 4) Refs. 4, 5, and 6. DB3 DB2 J

The low limit on increase in unidentified LEAKAGE assumes a failure mechanism ~~of intergranular stress corrosion cracking (IGSCC) that produces tight cracks. This flow increase limit is capable of providing an early warning of such deterioration~~. Insert ASA DB3

No applicable safety analysis assumes the total LEAKAGE limit. The total LEAKAGE limit considers RCS inventory makeup capability and drywell floor sump capacity.

RCS operational LEAKAGE satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (Ref. 7). X1 J

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, ~~being~~ because it is indicative of material degradation. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. PA2

(continued)

Revision J

INSERT ASA

that produces relatively tight cracks in piping and components. Intergranular stress corrosion cracking (IGSCC) would be typical of tight cracks on stainless steel piping and components susceptible to IGSCC (Refs. 8 and 9). Inspection programs (Refs. 10 and 11) are in place to detect cracking of piping and components.

5

BASES

ACTIONS

C.1 and C.2 (continued)

based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

The RCS LEAKAGE is monitored by a variety of instruments designed to provide alarms when LEAKAGE is indicated and to quantify the various types of LEAKAGE. Leakage detection instrumentation is discussed in more detail in the Bases for LCO 3.4.6 5, "RCS Leakage Detection Instrumentation." Sump level and flow rate are typically monitored to determine actual LEAKAGE rates; however, any method may be used to quantify LEAKAGE within the guidelines of Reference 5 8. In conjunction with alarms and other administrative controls, an 8 4 hour Frequency for this Surveillance is appropriate for identifying LEAKAGE and for tracking required trends (Ref. 6 9).

PA3 1 J

CLB2 DB2 1 J

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 30.~~

2. ~~GEAP-5620, April 1968.~~

3. ~~NUREG-76/067, October 1975.~~

4. ~~FSAR, Section [5.2.7.5.2].~~

5. ~~Regulatory Guide 1.45.~~

6. ~~Generic Letter 88-01, Supplement 1.~~

INSERT Ref

Revision J

AB2

Insert Ref

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. UFSAR, Section 16.6.
4. UFSAR, Section 4.10.
5. UFSAR, Section 16.3.
6. DRF-E31-00029-3(E), Summary of the Design of the Leak Detection System (LDS) for New York Power Authority, James A. FitzPatrick Nuclear Power Plant, November 1997.
7. 10 CFR 50.36(c)(2)(ii).
8. UFSAR, Section 4.10.3.4.
9. Generic Letter 88-01, NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping, US Nuclear Regulatory Commission, January 1988.
10. UFSAR, Section 16.4.
11. UFSAR, Section 16.5.14.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Operational LEAKAGE

BASES

BACKGROUND

The RCS includes systems and components that contain or transport the coolant to or from the reactor core. The pressure containing components of the RCS and the portions of connecting systems out to and including the isolation valves define the reactor coolant pressure boundary (RCPB). The joints of the RCPB components are welded or bolted. Some joints in ≤ 1 inch piping are also threaded.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. Limits on RCS operational LEAKAGE are required to ensure appropriate action is taken before the integrity of the RCPB is impaired. This LCO specifies the types and limits of LEAKAGE. This protects the RCS pressure boundary described in 10 CFR 50.2, 10 CFR 50.55a(c), and UFSAR, Section 16.6 (Refs. 1, 2, and 3).

The safety significance of RCS LEAKAGE from the RCPB varies widely depending on the source, rate, and duration. Therefore, detection of LEAKAGE in the drywell is necessary. Methods for quickly separating the identified LEAKAGE from the unidentified LEAKAGE are necessary to provide the operators quantitative information to permit them to take corrective action should a leak occur that is detrimental to the safety of the facility or the public.

A limited amount of leakage inside the drywell is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected and isolated from the primary containment atmosphere, if possible, so as not to mask RCS operational LEAKAGE detection.

This LCO deals with protection of the RCPB from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The allowable RCS operational LEAKAGE limits are based on the predicted and experimentally observed behavior of pipe cracks. The normally expected background LEAKAGE due to equipment design and the detection capability of the instrumentation for determining system LEAKAGE were also considered. The evidence from experiments suggests that, for LEAKAGE even greater than the specified unidentified LEAKAGE limits, the probability is small that the imperfection or crack associated with such LEAKAGE would grow rapidly.

The unidentified LEAKAGE flow limit allows time for corrective action before the RCPB could be significantly compromised. The 5 gpm limit is a small fraction of the calculated flow from a critical crack in the primary system piping. Crack behavior from experimental programs shows that leakage rates much greater than 5 gpm will precede crack instability (Refs. 4, 5, and 6). J

The low limit on increase in unidentified LEAKAGE assumes a failure mechanism that produces relatively tight cracks in piping and components. Intergranular stress corrosion cracking (IGSCC) would be typical of tight cracks on stainless steel piping and components susceptible to IGSCC (Refs. 8 and 9). Inspection programs (Refs. 10 and 11) are in place to detect cracking of piping and components. J

No applicable safety analysis assumes the total LEAKAGE limit. The total LEAKAGE limit considers RCS inventory makeup capability and drywell floor sump capacity.

RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 7). J

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, because it is indicative of material degradation. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

The RCS LEAKAGE is monitored by a variety of instruments designed to provide alarms when LEAKAGE is indicated and to quantify the various types of LEAKAGE. Leakage detection instrumentation is discussed in more detail in the Bases for LCO 3.4.5, "RCS Leakage Detection Instrumentation." Sump level and flow rate are typically monitored to determine actual LEAKAGE rates; however, any method may be used to quantify LEAKAGE within the guidelines of Reference 8. In conjunction with alarms and other administrative controls, a 4 hour Frequency for this Surveillance is appropriate for identifying LEAKAGE and for tracking required trends (Ref. 9).

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. UFSAR, Section 16.6.
4. UFSAR, Section 4.10.
5. UFSAR, Section 16.3.
6. DRF-E31-00029-3(E), Summary of the Design of the Leak Detection System (LDS) for New York Power Authority, James A. FitzPatrick Nuclear Power Plant, November 1997.
7. 10 CFR 50.36(c)(2)(ii).
8. UFSAR, Section 4.10.3.4.
9. Generic Letter 88-01, NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping, US Nuclear Regulatory Commission, January 1988.

(continued)

BASES

REFERENCES (continued)	10.	UFSAR, Section 16.4.
	11.	UFSAR, Section 16.5.14.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L5 (continued)

Detection Instrumentation, is to provide instrumentation requirements for early identification of unidentified leakage, the drywell equipment drain sump monitoring system requirements of CTS 3.6.D.4, 3.6.D.5, 4.6.D.4, Table 3.2-5, and Table 4.2-5 are proposed to be deleted. The drywell equipment drain sump monitoring system does not necessarily relate directly to the Leakage requirements (other means to quantify identified leakage are available, such as equipment drain sump pump-out times). Control of the availability of, and necessary compensatory activities if not available, for indications and monitoring instruments are addressed by plant operational procedures and policies. The requirement to demonstrate Leakage is within limits is still maintained in SR 3.4.4.1. As a result, the requirement for a means to quantify identified leakage is adequately addressed by the requirements of ITS 3.4.4 and associated SR 3.4.4.1. Therefore, explicit requirements for the drywell equipment drain sump monitoring system instrumentation are not required.

L6 A Note has been added to CTS 4.6.D.4 (Note to ITS 3.4.5 Surveillance Requirements) to allow a channel to be inoperable for up to 6 hours solely for performance of required Surveillances provided the other Leakage Detection System channel is OPERABLE. The 6 hour testing allowance has been granted by the NRC in Technical Specification amendment for Georgia Power Company's Hatch Unit 1 (Amendment 185) and Unit 2 (Amendment 125), in the ITS amendment for Washington Public Power Supply System Unit 2 (amendment 149), Nine Mile Point Unit 2 (Amendment 91, the ITS amendment), and LaSalle Units 1 and 2 (Amendments 147/133, respectively, the ITS amendments). The NRC has also granted this allowance in other topical reports for the Reactor Protection System, Emergency Core Cooling System, and Isolation System Instrumentation. The 6 hour testing allowance does not significantly reduce the probability of properly monitoring leakage since the other channel must be OPERABLE for this allowance to be used.

TECHNICAL CHANGES - RELOCATIONS

None

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Leakage Detection Instrumentation

5 PA1

BASES

BACKGROUND

~~GDC 30 of 10 CFR 50, Appendix A~~ The JAFNPP design basis (Ref. 1), requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. ~~Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.~~ DB1 J

Insert BKGD-1 DB2

~~Limits on LEAKAGE from the reactor coolant pressure boundary (RCPB) are required so that appropriate action can be taken before the integrity of the RCPB is impaired (Ref. 2).~~ J

Insert BKGD-2

Leakage detection systems for the RCS are provided to alert the operators when leakage rates above normal background levels are detected and also to supply quantitative measurement of leakage rates. The Bases for LCO 3.4.4, "RCS Operational LEAKAGE," discuss the limits on RCS LEAKAGE rates.

Systems for separating the LEAKAGE of an identified source from an unidentified source are necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action.

CLB1

LEAKAGE from the RCPB inside the drywell is detected by at least one of two ~~or three~~ independently monitored variables, such as sump pump flow ~~level changes~~ and drywell gaseous and particulate radioactivity levels. The primary means of quantifying LEAKAGE in the drywell is the drywell floor drain sump monitoring system.

The drywell floor drain sump monitoring system monitors the LEAKAGE collected in the floor drain sump. This unidentified LEAKAGE consists of LEAKAGE from control rod drives, valve flanges or packings, floor drains, the Reactor Building Closed Loop Cooling Water System, and drywell air cooling unit condensate drains, and any LEAKAGE not collected in the drywell equipment drain sump. The ~~primary containment~~ drywell floor drain sump has ~~transmitters~~ instrumentation that supply level indications in the ~~main~~ control room. PA2 PA2 PA2 PA2

The floor drain sump level ~~indicators~~ instrumentation ~~have~~ include switches that start and stop the sump pumps when required. A timer starts each time the sump is pumped down to the low level setpoint. PA2

(continued)

Revision J

DB2

Insert BKGD-1

Reliable means are provided to detect leakage from the reactor coolant pressure boundary (RCPB) before predetermined limits are exceeded (Refs. 2 and 3).

J

DB2

Insert BKGD-2

are established on abnormal leakage so that corrective action can be taken before unacceptable results occur (Ref. 4)

J

DB2 unless otherwise noted

In addition, the pump-out time is monitored and whenever the pump-out time exceeds a preset interval (indicating an increase in leak rate) an alarm annunciates in the control room

BASES

BACKGROUND (continued)

If the sump fills to the high level setpoint before the timer ends, an alarm sounds in the control room, indicating a LEAKAGE rate into the sump in excess of a preset limit.

Insert BKGD PA2

~~A flow indicator in the discharge line of the drywell floor drain sump pumps provides flow indication in the control room. The pumps can also be started from the control room.~~

drywell continuous atmospheric PA2 drywell

The ~~primary containment air~~ monitoring systems continuously monitor the ~~primary containment~~ atmosphere for airborne particulate and gaseous radioactivity. A sudden increase of radioactivity, which may be attributed to RCPB steam or reactor water LEAKAGE, is annunciated in the control room. The ~~primary containment~~ atmosphere particulate and gaseous radioactivity monitoring systems ~~are~~ is not capable of quantifying LEAKAGE rates, ~~but are sensitive enough to indicate increased LEAKAGE rates of 1 gpm within 1 hour. Larger changes in LEAKAGE rates are detected in proportionally shorter times (Ref. 3).~~ J

drywell PA2 DB2 Insert BKGD-3

~~[Condensate from four of the six primary containment coolers is routed to the primary containment floor drain sump and is monitored by a flow transmitter that provides indication and alarms in the control room. This primary containment air cooler condensate flow rate monitoring system serves as an added indicator, but not quantifier, of RCS unidentified LEAKAGE.]~~ DB3

APPLICABLE SAFETY ANALYSES

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. ~~4~~ 6 and ~~5~~ 7). Each of the leakage detection systems inside the drywell is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits and providing appropriate alarm or indication of excess LEAKAGE in the control room. DB4 J

DB2

A control room alarm or indication allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. ~~6~~ 6 and 7). Therefore, these actions provide adequate response 5 before a significant break in the RCPB can occur. DB4 J

(continued)

Revision J

DBL Insert BKGD

As the water which has been collected in the drywell floor drain sump is pumped out, the discharge flow is measured and total flow indicated by a flow integrator. The unidentified LEAKAGE and unidentified LEAKAGE increase are periodically calculated from this flow integrator. A flow recorder continually plots time versus discharge flow rate: an increase in leakage rate is also detectable by an increase in sump discharge flow time and an increased frequency in discharge flow cycles.

DBL Insert BKGD-3

. The sensitivity and response time of the system are a function of: location of the leak; amount of fission or corrosion product remaining in the atmosphere where they may be measured; plateout of these products in the sampling lines; effectiveness of the drywell coolers in reducing airborne concentrations; and the power level at the time of leakage occurrence (Ref. 3). The drywell continuous atmospheric particulate monitoring system is sufficiently sensitive to detect a reactor coolant leak of 1 gpm within 4 hours. The drywell continuous atmospheric gaseous monitoring system, however, will not alarm for reactor coolant leaks (since there is no retention factor for noble gases in the reactor coolant). The drywell continuous atmospheric gaseous monitoring system will respond only if the leak is in the steam portion of the RCPB. Larger changes in LEAKAGE rates can be detected in proportionally shorter times (Ref. 5). 5

BASES

APPLICABLE SAFETY ANALYSES (continued)	RCS leakage detection instrumentation satisfies Criterion 1 of ~~the NRC Policy Statement~~. 10 CFR 50.36 (c)(2)(ii) (Ref. 8) J
LCO PA3	The drywell floor drain sump monitoring system is required to quantify the unidentified LEAKAGE from the RCS. Thus, for the system to be considered OPERABLE, ~~either~~ the flow monitoring ~~or the sump level monitoring~~ portion of the system must be OPERABLE since this portion is capable of quantifying unidentified LEAKAGE from the RCS. CLB1 The other monitoring systems (one channel each of the drywell continuous atmospheric particulate and drywell continuous atmospheric gaseous monitoring systems) provide early alarms to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. With the leakage detection systems inoperable, monitoring for LEAKAGE in the RCPB is degraded.
APPLICABILITY	In MODES 1, 2, and 3, leakage detection systems are required to be OPERABLE to support LCO 3.4.4. This Applicability is consistent with that for LCO 3.4.4.

ACTIONS

A.1

PA2

With the drywell floor drain sump monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, the ~~primary containment~~ drywell continuous atmospheric ~~activity~~ monitor~~s~~ ~~and the primary containment air cooler condensate flow rate monitor~~ DB2 will provide indication of changes in leakage.

PA4

With the drywell floor drain sump monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every ~~8~~ 4 hours (SR 3.4.4.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still available. Required Action A.1 is modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the drywell floor drain sump monitoring system is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

(continued)

Revision J

X2 Insert SR NOTE

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, provided the other required instrumentation (the drywell floor drain sump monitoring system or drywell continuous atmospheric monitoring channel, as applicable) is OPERABLE. 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 probability of properly monitoring RCS leakage.

TA2 INSERT A

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.

DB4 Insert REF

REFERENCES

1. UFSAR, Section 16.6. J
2. UFSAR, Section 4.10.1. J
3. UFSAR, Section 4.10.3.4.
4. UFSAR, Section 4.10.2.3. J
5. JAF-CALC-PRM-03345, Rev. 0, March 2000.
6. UFSAR, Section 4.10.3.2.
7. UFSAR, Section 16.3.2.2. J
8. 10 CFR 50.36(c)(2)(ii). J

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

The JAFNPP design basis (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Reliable means are provided to detect leakage from the reactor coolant pressure boundary (RCPB) before predetermined limits are exceeded (Refs. 2 and 3).

Limits are established on abnormal leakage so that corrective action can be taken before unacceptable results occur (Ref. 4). Leakage detection systems for the RCS are provided to alert the operators when leakage rates above normal background levels are detected and also to supply quantitative measurement of leakage rates. The Bases for LCO 3.4.4, "RCS Operational LEAKAGE," discuss the limits on RCS LEAKAGE rates.

Systems for separating the LEAKAGE of an identified source from an unidentified source are necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action.

LEAKAGE from the RCPB inside the drywell is detected by at least one of two independently monitored variables, such as sump pump flow and drywell gaseous and particulate radioactivity levels. The primary means of quantifying LEAKAGE in the drywell is the drywell floor drain sump monitoring system.

The drywell floor drain sump monitoring system monitors the LEAKAGE collected in the floor drain sump. This unidentified LEAKAGE consists of LEAKAGE from control rod drives, valve flanges or packings, floor drains, the Reactor Building Closed Loop Cooling Water System, and drywell air cooling unit condensate drains, and any LEAKAGE not collected in the drywell equipment drain sump. The drywell floor drain sump has instrumentation that supply level indicators in the control room.

The floor drain sump level instrumentation include switches that start and stop the sump pumps where required. A timer starts each time the sump is pumped down to the low level setpoint. If the sump fills to the high level setpoint

(continued)

BASES

BACKGROUND (continued)

before the timer ends, an alarm sounds in the control room, indicting a LEAKAGE rate into the sump in excess of a preset limit. In addition, the pump-out time is monitored and whenever the pump-out time exceeds a preset interval (indicating an increase in leak rate) an alarm annunciates in the control room.

As the water which has been collected in the drywell floor drain sump is pumped out, the discharge flow is measured and total flow indicated by a flow integrator. The unidentified LEAKAGE and unidentified LEAKAGE increase are periodically calculated from this flow integrator. A flow recorder continually plots time versus discharge flow rate: an increase in leakage rate is also detectable by an increase in sump discharge flow time and an increased frequency in discharge flow cycles.

The drywell continuous atmospheric monitoring system continuously monitors the drywell atmosphere for airborne particulate and gaseous radioactivity. A sudden increase of radioactivity, which may be attributed to RCPB steam or reactor water LEAKAGE, is annunciated in the control room. The drywell atmosphere particulate and gaseous radioactivity monitoring system is not capable of quantifying LEAKAGE rates. The sensitivity and response time of the system are a function of: location of the leak; amount of fission or corrosion product remaining in the atmosphere where they may be measured; plateout of these products in the sampling lines; effectiveness of the drywell coolers in reducing airborne concentrations; and the power level at the time of leakage occurrence (Ref. 3). The drywell continuous atmospheric particulate monitoring system is sufficiently sensitive to detect a reactor coolant leak of 1 gpm within 4 hours. The drywell continuous atmospheric gaseous monitoring system, however, will not alarm for reactor coolant leaks (since there is no retention factor for noble gases in the reactor coolant). The drywell continuous atmospheric gaseous monitoring system will respond only if the leak is in the steam portion of the RCPB. Larger changes in LEAKAGE rates can be detected in proportionally shorter times (Ref. 5).

APPLICABLE SAFETY ANALYSES

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. 6

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

and 7). Each of the leakage detection systems inside the drywell is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits and providing appropriate alarm or indication of excess LEAKAGE in the control room.

A control room alarm or indication allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Refs. 6 and 7). Therefore, these actions provide adequate response before a significant break in the RCPB can occur. RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii) (Ref. 8).

LCO

The drywell floor drain sump monitoring system is required to quantify the unidentified LEAKAGE from the RCS. Thus, for the system to be considered OPERABLE, the flow monitoring portion of the system must be OPERABLE since this portion is capable of quantifying unidentified LEAKAGE from the RCS. The other monitoring systems (one channel each of the drywell continuous atmospheric particulate and drywell continuous atmospheric gaseous monitoring systems) provide early alarms to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. With the leakage detection systems inoperable, monitoring for LEAKAGE in the RCPB is degraded.

APPLICABILITY

In MODES 1, 2, and 3, leakage detection systems are required to be OPERABLE to support LCO 3.4.4. This Applicability is consistent with that for LCO 3.4.4.

ACTIONS

A.1

With the drywell floor drain sump monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, the drywell continuous atmospheric monitors will provide indication of changes in leakage.

(continued)

BASES

ACTIONS

A.1 (continued)

With the drywell floor drain sump monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every 4 hours (SR 3.4.4.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still available. Required Action A.1 is modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the drywell floor drain sump monitoring system is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

B.1

With one required drywell continuous atmospheric monitoring channel inoperable, SR 3.4.5.1 must be performed every 8 hours for the remaining OPERABLE drywell continuous atmospheric monitoring channel to provide periodic information of activity in the drywell at a more frequent interval than the routine Frequency of SR 3.4.5.1. The 8 hour interval provides periodic information that is adequate to detect LEAKAGE and recognizes that other forms of leakage detection are available. However, this Required Action is modified by a Note that allows this action to be not applicable if both drywell continuous atmospheric monitoring systems are inoperable. Consistent with SR 3.0.1, Surveillances are not required to be performed on inoperable equipment. J

C.1 and C.2

With both required gaseous and particulate drywell continuous atmospheric monitoring channels inoperable, grab samples of the drywell atmosphere must be taken and analyzed to provide periodic leakage information. Provided a sample is obtained and analyzed once every 12 hours, the plant may be operated for up to 30 days to allow restoration of at least one of the two monitors.

The 12 hour interval provides periodic information that is adequate to detect LEAKAGE. The 30 day Completion Time for restoration recognizes that at least one other form of leakage detection is available.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

The Required Actions are modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when both the gaseous and particulate drywell continuous atmospheric monitoring systems are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

D.1 and D.2

If any Required Action and associated Completion Time of Condition A or B cannot be 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 and MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the actions in an orderly manner and without challenging plant systems.

E.1

With all required monitors inoperable, no required automatic means of monitoring LEAKAGE are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

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, provided the other required instrumentation (the drywell floor drain sump monitoring system or drywell continuous atmospheric monitoring channel, as applicable) is OPERABLE. 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 probability of properly monitoring RCS leakage.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.1

This SR is for the performance of a CHANNEL CHECK of the required drywell continuous atmospheric monitoring channels. The check gives reasonable confidence that the channels are operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.5.2

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies the alarm setpoint and relative accuracy of the instrument channel. 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. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.5.3

This SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument channel. The Frequency is 92 days and operating experience has proven this Frequency is acceptable.

REFERENCES

1. UFSAR, Section 16.6.
2. UFSAR, Section 4.10.1.
3. UFSAR, Section 4.10.3.4.
4. UFSAR, Section 4.10.2.3.

(continued)

BASES

REFERENCES (continued)

5. JAF-CALC-PRM-03345, Rev. 0, March 2000.
6. UFSAR, Section 4.10.3.2.
7. UFSAR, Section 16.3.2.2.
8. 10 CFR 50.36(c)(2)(ii).

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.~~8~~.1 7 PA1	-------------------NOTE------------------- Not required to be met until 2 hours after reactor steam dome pressure is ~~≥~~ less than the RHR cut in permissive pressure. PA3 PA2 --- Verify ~~one~~ each required RHR shutdown cooling subsystem ~~or recirculation pump is operating~~ manual, power operated, and automatic valve in the flow path that is not locked, sealed or otherwise secured in position, is in the correct position, or can be aligned to the correct position. CLB1	~~12 hours~~ 31 days CLB1

| J

Revision J

(and two RHR service water pumps) PA3

BASES

LCO (continued)

PA3 RHR (and one RHR service water pump) PA3 RHR

common discharge piping. Thus, to meet the LCO, both pumps in one loop or one pump in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned ~~(remote or local)~~ in the shutdown cooling mode for removal of decay heat. In MODE 3, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

PA4 from the control room or locally

~~Note 1 permits both RHR shutdown cooling subsystems to be shut down for a period of 2 hours in an 8 hour period.~~ The Note ~~2~~ allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR ~~System~~ in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring ~~RHR flow interruption and~~ loss of redundancy.

CLB1 shutdown cooling subsystem X2 PA3 shutdown cooling PA3 5

(Function 6.a of LCO 3.3.6.1, "Primary Containment Isolation Instrumentation")

APPLICABILITY

PA3 PA4 is required to be OPERABLE so that it

In MODE 3 with reactor steam dome pressure below ~~(~~the RHR cut in permissive pressure~~)~~ (i.e., the actual pressure at which the interlock resets) the RHR System ~~may be operated~~ in the shutdown cooling mode to remove decay heat to reduce or maintain coolant temperature. Otherwise, a recirculation pump is ~~required to be in operation~~. to circulate coolant to provide for temperature monitoring

shutdown cooling suction valve isolation logic PA3 CLB1

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to ~~(~~the RHR cut in permissive pressure~~)~~, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut in permissive pressure is typically accomplished by condensing

normally in operation CLB1 PA4 PA3

(continued)

Revision 5

PA1

BASES

ACTIONS

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

CLB1

and is modified such that the 1 hour is applicable separately for each occurrence involving a loss of coolant circulation. Furthermore, verification of the functioning of the alternate method must be reconfirmed every 12 hours thereafter. This will provide assurance of continued temperature monitoring capability.

During the period when the reactor coolant is being circulated by an alternate method (other than by the required RHR shutdown cooling subsystem or recirculation pump), the reactor coolant temperature and pressure must be periodically monitored to ensure proper function of the alternate method. The once per hour Completion Time is deemed appropriate.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1 7 PA1

Insert SR 3.4.7.1

CLB1

This Surveillance verifies that one RHR shutdown cooling subsystem or recirculation pump is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.

CLB1

subsystem

OPERABILITY

Verify

This Surveillance is modified by a Note allowing sufficient time to ~~align the~~ RHR ~~system for~~ shutdown cooling operation after clearing the pressure interlock that isolates the system~~, or for placing a recirculation pump in operation~~. The Note takes exception to the requirements of the Surveillance being met (i.e., ~~forced coolant circulation is~~ not required for this initial 2 hour period), which also allows entry into the Applicability of this Specification in accordance with SR 3.0.4 since the Surveillance will not be "not met" at the time of entry into the Applicability.

valves are aligned or can be aligned is

REFERENCES ~~None.~~ 2. 10 CFR 50.36(c)(2)(ii) X1

1. UFSAR, chapter 14.

DB1

1J

Revision J

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	-------------------NOTE------------------- Not required to be met until 2 hours after reactor steam dome pressure is less than the RHR cut in permissive pressure. -- Verify each required RHR shutdown cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position, or can be aligned to the correct position.	31 days

BASES

LCO (continued)

each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (from the control room or locally) in the shutdown cooling mode for removal of decay heat. In MODE 3, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

The Note allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR shutdown cooling subsystem in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR shutdown cooling subsystems or other operations requiring loss of redundancy.

APPLICABILITY

In MODE 3 with reactor steam dome pressure below the RHR cut-in permissive pressure (i.e., the actual pressure at which the shutdown cooling suction valve isolation logic interlock resets (Function 6.a of LCO 3.3.6.1, "Primary Containment Isolation Instrumentation")) the RHR System is required to be OPERABLE so that it may be operated in the shutdown cooling mode to remove decay heat to reduce or maintain coolant temperature. Otherwise, a recirculation pump is normally in operation to circulate coolant to provide for temperature monitoring.

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR cut-in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No RHR shutdown cooling subsystem in operation. AND No recirculation pump in operation.	B.1	Verify reactor coolant circulating by an alternate method.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	AND B.2	Monitor reactor coolant temperature.	Once per hour

CLB1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1 8 PA1	Verify ~~one~~ each RHR shutdown cooling subsystem ~~or recirculation pump is operating~~.	~~12 hours~~ 31 days

CLB1

manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position, or can be aligned to the correct position.

J

Revision J

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown

8 PA1

BASES

the decay heat must be removed for PA3

BACKGROUND PA2 212

Irradiated fuel in the shutdown reactor core generates heat during the decay of fission products and increases the temperature of the reactor coolant. This decay heat must be removed to maintain the temperature of the reactor coolant ≤ 200°F. ~~This decay heat removal is~~ in preparation for performing refueling ~~or maintenance~~ operations, or for keeping the reactor in the Cold Shutdown condition.

Maintaining PA3

The two redundant, manually controlled shutdown cooling subsystems of the RHR System provide decay heat removal. Each loop consists of two motor driven pumps, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same recirculation loop. Each pump discharges the reactor coolant, after circulation through the respective heat exchanger, to the reactor via the associated recirculation loop. The RHR heat exchangers transfer heat to the RHR Service Water System.

(loops) PA3 reactor water PA4 PA6 a DB2 PA4 reactor water (Ref. 1)

APPLICABLE SAFETY ANALYSES

Decay heat removal by operation of the RHR System in the shutdown cooling mode is not required for mitigation of any event or accident evaluated in the safety analyses. Decay heat removal is, however, an important safety function that must be accomplished or core damage could result. ~~Although the RHR Shutdown Cooling System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as a significant contributor to risk reduction. Therefore, the RHR Shutdown Cooling System is retained as a Technical Specification.~~

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

LCO DB1

Two RHR shutdown cooling subsystems are required to be OPERABLE~~, and when no recirculation pump is in operation, one RHR shutdown cooling subsystem must be in operation~~. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, and the associated piping and valves. The two subsystems have a common suction source and are allowed to have a common heat exchanger and common discharge piping. Thus, to meet the LCO, both pumps

one or two RHR service water pumps providing water to the heat exchanger, as required for temperature control,

CLB1 PA3 RHR 1 J

(continued)

Typ. All Pages

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(and two RHR service water pumps) PA3 (and one RHR service water pump) J

BASES

RHR

LCO (continued)

in one loop or one pump in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. In MODE 4, the RHR cross tie valve (2E11-F010) may be opened to allow pumps in one loop to discharge through the opposite recirculation loop to make a complete subsystem. Additionally, each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (~~remote or local~~) in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

DB1 valves (10 MOV-20 and 10 RHR-09)

From the control room or locally PA4

CLB1

~~Note 1 permits both RHR shutdown cooling subsystems to be shut down for a period of 2 hours in an 8 hour period.~~ Note ~~2~~ allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR ~~System~~ in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring ~~RHR flow interruption and~~ loss of redundancy.

The CLB1

shutdown cooling subsystems PA3

shutdown cooling PA2

CLB1

hormally in operation PA3

is required to be OPERABLE so that it PA2

APPLICABILITY

In MODE 4, the RHR ~~Shutdown Cooling~~ System may be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below ~~200~~ 212°F. Otherwise, a recirculation pump is ~~required to be in operation~~.

to circulate coolant to provide for temperature monitoring CLB1 PA3

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR cut in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut in permissive pressure is typically accomplished by condensing

(continued)

Revision J

PA1

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 8 PA1

CLB1

Insert SR 3.4.8.1

CLB1

~~This Surveillance verifies that one RHR shutdown cooling subsystem or recirculation pump is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.~~

REFERENCES

None. 2. 10 CFR 50.36 (c)(2)(ii). X1

1. UFSAR, Chapter 14.

DB2

J

Revision J

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify each RHR shutdown cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position, or can be aligned to the correct position.	31 days

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 Residual Heat Removal (RHR) Shutdown Cooling System – Cold Shutdown

BASES

BACKGROUND

Irradiated fuel in the shutdown reactor core generates heat during the decay of fission products and increases the temperature of the reactor coolant. This decay heat must be removed to maintain the temperature of the reactor coolant ≤ 212°F in preparation for performing refueling operations, or the decay heat must be removed for maintaining the reactor in the Cold Shutdown condition.

The two redundant, manually controlled shutdown cooling subsystems (loops) of the RHR System provide decay heat removal. Each loop consists of two motor driven pumps, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same reactor water recirculation loop. Each pump discharges the reactor coolant, after circulation through the respective heat exchanger, to the reactor via a reactor water recirculation loop. The RHR heat exchangers transfer heat to the RHR Service Water System.

APPLICABLE SAFETY ANALYSES

Decay heat removal by operation of the RHR System in the shutdown cooling mode is not required for mitigation of any event or accident evaluated in the safety analyses (Ref. 1). Decay heat removal is, however, an important safety function that must be accomplished or core damage could result. The RHR Shutdown Cooling System meets Criterion 4 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

LCO

Two RHR shutdown cooling subsystems are required to be OPERABLE. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, one or two RHR service water pumps providing water to the heat exchanger, as required for temperature control, and the associated piping and valves. The two subsystems have a common suction source and are allowed to have a common heat exchanger and common discharge piping. Thus, to meet the LCO, both RHR pumps (and two RHR service water pumps) in one

(continued)

BASES

LCO (continued)

loop or one RHR pump (and one RHR service water pump) in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. In MODE 4, the RHR cross tie valves (10MOV-20 and 10RHR-09) may be opened to allow pumps in one loop to discharge through the opposite recirculation loop to make a complete subsystem. Additionally, each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (from the control room or locally) in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required. | J

The Note allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR shutdown cooling subsystems in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR shutdown cooling subsystems or other operations requiring loss of redundancy.

APPLICABILITY

In MODE 4, the RHR System is required to be OPERABLE so that it may be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below 212°F. Otherwise, a recirculation pump is normally in operation to circulate coolant to provide for temperature monitoring.

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR cut in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2

(continued)

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME
[M4]	C. ---------NOTE--------- Required Action C.2 shall be completed if this Condition is entered. --------------------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
		AND		
		C.2	Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3.

SURVEILLANCE REQUIREMENTS

		SURVEILLANCE	FREQUENCY
[4.6.A.2] [4.6.A.3] [3.6.A.2] [3.6.A.3] [3.6.A.4] INSERT SR-1	SR 3.4.~~10~~ 9 .1 PA1	-------------------NOTE------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. --- ~~Verify~~ a. RCS pressure, and RCS temperature ~~and RCS heatup and cooldown~~ ~~rates~~ are within the limits specified in ~~the PTLR.~~ Figure 3.4.9-1 or Figure 3.4.9-2, as applicable; and	30 minutes CLB1
[4.6.A.4]	SR 3.4.~~10~~ 9 .2 PA1	Verify RCS pressure and RCS temperature are within the criticality limits specified in ~~the PTLR.~~ Figure 3.4.9-1 or Figure 3.4.9-2, as applicable CLB1	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

J

(continued)

Revision J

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
[4.6.A.6.a] [L1] [3.6.A.6.c] PA1 X1 Insert SR 3.4.9.4 [L2]	SR 3.4.~~10~~ 9.3 X1 ------NOTE------ 1. Only required to be met in MODES 1, 2, 3, and 4 ~~with reactor steam dome pressure ≥ 25 psig~~ X1 DB1 ------ Insert Note 2 X1 ------ Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ~~within the limits specified in the PTLR~~ ≤ 145 °F CLB1.	Once within 15 minutes prior to each startup of a recirculation pump during recirculation pump startup. TA1
[4.6.A.6.b] [4.6.A.6.c] [L1] PA1	SR 3.4.~~10~~ 9.~~4~~ 5 X1 ------NOTE------ Only required to be met in MODES 1, 2, 3, and ~~4~~. ------ Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ~~within the limits specified in the PTLR~~ ≤ 50 °F CLB1.	Once within 15 minutes prior to each startup of a recirculation pump during recirculation pump startup. TA1
[4.6.A.1.c] [3.6.A.1] PA1	SR 3.4.~~10~~ 9.~~5~~ 6 X1 ------NOTE------ Only required to be performed when tensioning the reactor vessel head bolting studs. ------ Verify, when the reactor vessel head bolting studs are under tension, reactor vessel flange and head flange temperatures are ~~within the limits specified in the PTLR~~ ≥ 90 °F CLB1.	30 minutes

(continued)

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X1 Insert Note 2

2. Not required to be performed if SR 3.4.9.4 is satisfied. 3

X1 Insert SR 3.4.9.4

SR 3.4.9.4 ------------------NOTES------------------- 1. Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. 2. Not required to be met if SR 3.4.9.3 is satisfied. --- Verify the active recirculation pump flow exceeds 40% of rated pump flow or the active recirculation pump has been operating below 40% rated flow for a period no longer than 30 minutes.	Once within 15 minutes prior to each startup of a recirculation pump

, when the reactor vessel head bolting studs are under tension, CLB1

J

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
[4.6.A.1.b] SR 3.4.~~10~~.~~6~~ 9 7 X1 PA1	-------------------NOTE------------------- Not required to be performed until 30 minutes after RCS temperature ≤ ~~80~~ 100 °F ~~in MODE 4~~. CLB1 with any reactor vessel stud tensioned --- Verify, when the reactor vessel head bolting studs are under tension, reactor vessel flange and head flange temperatures are ~~within the limits specified in the PTLR~~ ≥90°F. CLB1	30 minutes
[4.6.A.1.a] SR 3.4.~~10~~.~~0~~ 9 8 X1 PA1	-------------------NOTE------------------- Not required to be performed until 12 hours after RCS temperature ≤ ~~100~~ 120 °F ~~in MODE 4~~. CLB1 head bolting --- Verify, when the reactor vessel head bolting studs are under tension, reactor vessel flange and head flange temperatures are ~~within the limits specified in the PTLR~~ ≥90°F. CLB1	12 hours

INSERT Figures 3.4.9-1 and 3.4.9-2 CLB1

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JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 JAFNPP has not developed the "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)". References to limits in the PTLR are replaced with current requirements. In addition, consistent with current licensing basis (CTS 3.6.A.1), the reactor vessel and head flange minimum temperature is only required to be met when the reactor vessel head bolting studs are under tension. Therefore, SR 3.4.9.6, SR 3.4.9.7, and SR 3.4.9.8 have been modified to reflect this condition. J

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 NUREG-1433 Specification 3.4.5, "RCS Pressure Isolation Valve (PIV) Leakage", is not incorporated in ITS. Subsequent ITS Specifications and Bases have been renumbered accordingly.

PA2 Editorial changes have been made to achieve consistency with the Writer's Guide for the Restructured Technical Specifications.

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 The bracketed allowance has been deleted since it does not apply to JAFNPP.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 35, Revision 0, have been incorporated.

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

X1 ITS SR 3.4.9.4 has been added to the requirements of ISTS 3.4.10 (ITS 3.4.9) to allow an alternative to the requirements of ITS SR 3.4.9.3. This Surveillance has been added to the CTS in accordance with L1. A Note 2 was added to SR 3.4.9.3 which allows the option to perform SR 3.4.9.4. In addition, subsequent Surveillances have been renumbered, as required.

BASES

APPLICABLE SAFETY ANALYSES (continued)

limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (Ref. 9). X1

LCO

The elements of this LCO are:

CLB1

INSERT LCO-1

a. RCS pressure and temperature, ~~and heatup or cooldown rate~~ are within the limits specified in ~~the PTLR, during RCS heatup, cooldown, and inservice leak and hydrostatic testing;~~ Figure 3.4.9-1 or Figure 3.4.9-2, as applicable. 1J

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is ~~within the limits of the PTLR~~ ≤ 145°F during recirculation pump startup~~, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow~~; CLB1 DB1

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel ~~meets the limit of the PTLR~~ is ≤ 50°F during recirculation pump startup~~, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow~~; CLB1 DB1

d. RCS pressure and temperature are within the ~~criticality~~ limits specified in ~~the PTLR~~ Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, prior to achieving criticality; and

e. The reactor vessel flange and the head flange temperatures are ~~within the limits of the PTLR~~ ≥ 90°F when tensioning the reactor vessel head bolting studs and when any stud is tensioned. CLB1 PA2

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

~~The rate of change of temperature limits control~~ The limits on the rate of change of RCS temperature, influenced by RCS flow and RCS stratification, control the thermal gradient through the vessel wall ~~and~~ and are used as inputs for calculating the heatup, cooldown, and inservice leakage and hydrostatic testing P/T limit curves. For this reason, both RCS temperature and RPV metal temperatures PA2 Thus, the LCO for the

(continued)

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.10.2 (9) PA1

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

X2 (SR 3.4.10.3 ~~and~~ SR 3.4.10.4 (9) PA1 (9) , SR 3.4.9.5 X2

Differential temperatures within the ~~applicable~~ PTLR limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 8 10) are satisfied. specified CLB1 DB2

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start. X2

INSERT SR 3.4.9.3-1 PA4

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.10.4 (9) (5) is to compare the temperatures of the operating recirculation loop and the idle loop. PA1 X2

TA1

during a recirculation pump startup since this is when the stresses occur

SR 3.4.10.3 (9) , SR 3.4.9.4, and SR 3.4.9.5 have been ~~has been~~ modified by a Note that requires the Surveillance to be ~~performed~~ only in MODES 1, 2, 3, and 4 with reactor steam dome pressure ≥ 25 psig. In MODE 5, the overall stress on limiting components is lower. Therefore, ΔT limits are not required. PA4 X2 Mef J PA4

INSERT Note 2 X2

SR 3.4.10.5, SR 3.4.10.6, and SR 3.4.10.7 (9) (9) (9) PA1 (6) (7) (8) X2

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits

(continued)

Revision J

X2 PA4 Insert SR 3.4.9.3-1

Compliance with the temperature differential requirement in SR 3.4.9.3 is demonstrated by comparing the bottom head drain line temperature to the reactor vessel steam dome saturation temperature. SR 3.4.9.4 requires the verification that the active recirculation loop flow exceeds 40% of rated drive flow or the active loop has been operating below 40% rated flow for a period no longer than 30 minutes. As specified in Reference 11 and 12, the alternative verification of SR 3.4.9.4 will ensure the temperature differential of SR 3.4.9.3 is met.

X2 Insert Note 2

SR 3.4.9.3 is modified by a second Note, which clarifies that the SR does not have to be performed if SR 3.4.9.4 is satisfied. This is acceptable since References 11 and 12 demonstrate that SR 3.4.9.4 is an acceptable alternative. In addition, SR 3.4.9.4 is modified by a second Note, which clarifies that the SR does not have to be met if SR 3.4.9.3 is satisfied. This is acceptable since SR 3.4.9.3 directly verifies the stratification limit is met.

PA1

BASES

SURVEILLANCE REQUIREMENTS SR 3.4.10.5, SR 3.4.10.6, and SR 3.4.10.7 (continued)

9 9 9 6 7 8 X2

during system heatup and cooldown. However, operations ~~approaching MODE 4 from MODE 5 and in MODE 4~~ When any reactor vessel stud is tensioned with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

The flange temperatures must be verified to be above the limits within 30 minutes before and while tensioning the reactor vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When ~~in MODE 4~~ any reactor vessel stud is tensioned with RCS temperature ≤ ~~80~~ 100°F, 30 minute checks of the flange temperatures are required because of the reduced margin to the limits. When ~~in MODE 4~~ any reactor vessel stud is tensioned with RCS temperature ≤ ~~100~~ 120°F, monitoring of the flange temperature is required every 12 hours to ensure the temperature is within ~~the~~ specified limits ~~specified in the PTLR~~. PA2 J CLB1 CLB1

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

INSERT SR 3.4.9.6 PA4

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. ASTM E 185-82, July 1982.
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials, May 1988.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
7. ~~NEDO-21778-A, December 1978.~~ DB2 PA3
INSERT Ref-1 DB2 J J
~~8.~~ 10 11. FSAR, Section ~~15.1.26~~ 14.5.7.1. DB2 DB3 X1 X2

Insert Ref-2

Revision J

Insert SR 3.4.9.6 PA4

SR 3.4.9.6 is modified by a Note which requires the SR to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.9.7 is modified by a Note which states that the SR is not required to be performed until 30 minutes after RCS temperature is ≤ 100°F in MODE 4. SR 3.4.9.8 is modified by a Note which states that the SR is not required to be performed until 12 hours after RCS temperature is ≤ 120°F in MODE 4. These Notes are necessary to specify when the reactor vessel flange and head flange temperatures are required to be within specified limits.

Insert Ref-1 PA2

7. GE-NE-B1100732-01, Revision 1, Plant FitzPatrick RPV Surveillance Materials Testing and Analysis of 120° Capsule at 13.4 EFPY, February 1998, including Errata and Addenda Sheets dated June 17, 1999 and December 3, 1999. J

8. Letter from Guy Vissing (NRC) to James Knubel (NYPA), Issuance of Amendment No. 258 to James A. FitzPatrick Nuclear Power Plant, November 29,1999.

9. 10 CFR 50.36(c)(2)(ii).

Insert Ref-2 X2

11. GE-NE-208-04-1292, Evaluation of Idle Recirculation Loop Restart Without Vessel Bottom Temperature Indication for FitzPatrick Nuclear Power Plant, December 1992.

12. JAF-RPT-RWR-02076, Verification of Alternative Operating Conditions for Idle Recirculation Loop Restart Without Vessel Bottom Temperature Indication, June 25, 1995. J

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	-----------------NOTE------------------ Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. --------------------------------------- Verify: a. RCS pressure and RCS temperature are within the limits specified in Figure 3.4.9-1 or Figure 3.4.9-2, as applicable; and b. RCS temperature change averaged over a one hour period is: 1. ≤ 100°F when the RCS pressure and RCS temperature are on or to the right of curve C of Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, during inservice leak and hydrostatic testing; 2. ≤ 20°F when the RCS pressure and RCS temperature are to the left of curve C of Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, during inservice leak and hydrostatic testing; and 3. ≤ 100°F during other heatup and cooldown operations.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.5	-------------------NOTE------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. -- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ≤ 50°F.	 Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.6	-------------------NOTE------------------- Only required to be performed when tensioning the reactor vessel head bolting studs. -- Verify, when the reactor vessel head bolting studs are under tension, reactor vessel flange and head flange temperatures are ≥ 90°F.	 30 minutes
SR 3.4.9.7	-------------------NOTE------------------- Not required to be performed until 30 minutes after RCS temperature ≤ 100°F with any reactor vessel head bolting stud tensioned. -- Verify, when the reactor vessel head bolting studs are under tension, reactor vessel flange and head flange temperatures are ≥ 90°F.	 30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.8	------------------NOTE------------------ Not required to be performed until 12 hours after RCS temperature ≤ 120°F with any reactor vessel head bolting stud tensioned. -- Verify, when the reactor vessel head bolting studs are under tension, reactor vessel flange and head flange temperatures are ≥ 90°F.	12 hours

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.2 (continued)

withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

SR 3.4.9.3, SR 3.4.9.4, and SR 3.4.9.5

Differential temperatures within the specified limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 10) are satisfied.

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

Compliance with the temperature differential requirement in SR 3.4.9.3 is demonstrated by comparing the bottom head drain line temperature to the reactor vessel steam dome saturation temperature. SR 3.4.9.4 requires the verification that the active recirculation pump flow exceeds 40% of rated pump flow or the active recirculation pump has been operating below 40% rated flow for a period no longer than 30 minutes. As specified in Reference 11 and 12, the alternative verification of SR 3.4.9.4 will ensure the temperature differential of SR 3.4.9.3 is met.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.5 is to compare the temperatures of the operating recirculation loop and the idle loop.

SR 3.4.9.3, SR 3.4.9.4, and SR 3.4.9.5 have been modified by a Note that requires the Surveillance to be met only in MODES 1, 2, 3, and 4 during a recirculation pump startup since this is when the stresses occur. In MODE 5, the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.3, SR 3.4.9.4 and SR 3.4.9.5 (continued)

overall stress on limiting components is lower. Therefore, ΔT limits are not required. SR 3.4.9.3 is modified by a second Note, which clarifies that the SR does not have to be performed if SR 3.4.9.4 is satisfied. This is acceptable since References 11 and 12 demonstrate that SR 3.4.9.4 is an acceptable alternative. In addition, SR 3.4.9.4 is modified by a second Note, which clarifies that the SR does not have to be met if SR 3.4.9.3 is satisfied. This is acceptable since SR 3.4.9.3 directly verifies the stratification limit is met.

SR 3.4.9.6, SR 3.4.9.7, and SR 3.4.9.8

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits during system heatup and cooldown. However, operations when any reactor vessel head bolting stud is tensioned with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

The flange temperatures must be verified to be above the limits within 30 minutes before and while tensioning the reactor vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When any reactor vessel head bolting stud is tensioned with RCS temperature ≤ 100°F, 30 minute checks of the flange temperatures are required because of the reduced margin to the limits. When any reactor vessel head bolting stud is tensioned with RCS temperature ≤ 120°F, monitoring of the flange temperature is required every 12 hours to ensure the temperature is within specified limits.

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

SR 3.4.9.6 is modified by a Note which requires the SR to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.9.7 is modified by a Note which states that the SR is not required to be performed until 30 minutes after RCS temperature is ≤ 100°F in MODE 4. SR 3.4.9.8 is modified by a Note which states that the SR is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.6, SR 3.4.9.7, and SR 3.4.9.8 (continued)

not required to be performed until 12 hours after RCS temperature is ≤ 120°F in MODE 4. These Notes are necessary to specify when the reactor vessel flange and head flange temperatures are required to be within specified limits.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. ASTM E 185-82, July 1982.
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials, May 1988.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
7. GE-NE-B1100732-01, Revision 1, Plant FitzPatrick RPV Surveillance Materials Testing and Analysis of 120° Capsule at 13.4 EFPY, February 1998, including Errata and Addenda Sheets dated June 17, 1999 and December 3, 1999.
8. Letter from Guy Vissing (NRC) to James Knubel (NYPA), Issuance of Amendment No. 258 to James A. FitzPatrick Nuclear Power Plant, November 29, 1999.
9. 10 CFR 50.36(c)(2)(ii).
10. UFSAR, Section 14.5.7.2.
11. GE-NE-208-04-1292, Evaluation of Idle Recirculation Loop Restart Without Vessel Bottom Temperature Indication for FitzPatrick Nuclear Power Plant, December 1992.
12. JAF-RPT-RWR-02076, Verification of Alternative Operating Conditions for Idle Recirculation Loop Restart Without Vessel Bottom Temperature Indication, June 25, 1995.

SUMMARY OF CHANGES TO ITS SECTION 3.5 - REVISION J

Source of Change	Summary of Change	Affected Pages
Retyped ITS typographical errors	Minor typographical errors in the retyped ITS have been corrected to be consistent with the NUREG markup. (The word "the" has been added to the last line of SR 3.5.1.10 Note 1; and the words "reactor steam dome pressure" has been changed to "reactor steam pressure" in SR 3.5.3.6 Note 1.)	**Specification 3.5.1** Retyped ITS p 3.5-6 **Specification 3.5.3** Retyped ITS p 3.5-15
NUREG ITS markup errors	Minor NUREG markup errors have been corrected to be consistent with the retyped ITS. (An extra space has been deleted between the word "inoperable" and the period in ITS 3.5.1 Insert ACTION A; and the word "valve" has been added to the first line of SR 3.5.1.12.)	**Specification 3.5.1** NUREG ITS markup p Insert page 3.5-1 and Insert page 3.5-6
Retyped ITS Bases typographical errors	Minor typographical errors in the retyped ITS Bases have been corrected to be consistent with the NUREG Bases markup. (The words "Design Basis" have been decapitalized in the ITS 3.5.1 LCO section; the words "Design Basis" have been decapitalized in the ITS 3.5.1 Actions C.1 and C.2 section, Actions D.1 and D.2 section, and Actions F.1 and F.2 section; the words "is inoperable" have been moved to after the words "LPCI subsystems" in ITS 3.5.1 Actions D.1 and D.2 section and Actions F.1 and F.2 section; the words "as specified in Reference" have been deleted in the SR 3.5.1.12 section; two paragraphs have been combined into one paragraph and the words "recommendation" has been changed to "recommendations" in the SR 3.4.1.13 section; the words "Design Basis" have been decapitalized in the ITS 3.5.2 ASA section; the words "within 1 hour" have been changed to "immediately" in the ITS 3.5.3 Actions A.1 and A.2 section; and the word "systemthroughout" has been changed to "system throughout" in the SR 3.5.3.5 section.)	**Specification 3.5.1** Retyped ITS Bases p B 3.5-5, B 3.5-7, B 3.5-8, B 3.5-9, B 3.5-16, and B 3.5-17 **Specification 3.5.2** Retyped ITS Bases p B 3.5-20 **Specification 3.5.3** Retyped ITS Bases p B 3.5-28 and B 3.5-31
NUREG Bases markup errors	Minor NUREG Bases markup errors have been corrected to be consistent with the retyped ITS Bases. (Parentheses has been placed around the words "Ref. 11"; a period has been added to ITS 3.5.1 Reference 11; a period has been added to the ITS 3.5.2 LCO section; a period has been added to ITS 3.5.2 Reference 2; a comma has been added to the ITS 3.5.3 Background section; the redundant term "psig" has been deleted in the SR 3.5.3.3 and SR 3.5.3.4 section; and periods have been added to ITS 3.5.3 References 1, 2, and 3.)	**Specification 3.5.1** NUREG Bases markup p B 3.5-5 and B 3.5-16 **Specification 3.5.2** NUREG Bases markup p B 3.5-18 and B 3.5-22 **Specification 3.5.3** NUREG Bases markup p B 3.5-23, B 3.5-27, and B 3.5-29
Typographical errors	Minor typographical errors have been corrected in the Discussion of Changes, NUREG Bases markups, and the retyped ITS Bases. (The term "SR 3.5.1.5" in ITS 3.5.1 DOC M10 has been changed to "SR 3.5.1.6"; a comma has been deleted in the ITS 3.5.1 Bases Actions F.1 and F.2 section; and dashes have been added between the words "Loss of Coolant" in the title of ITS 3.5.1 Bases Reference 5.)	**Specification 3.5.1** DOC M10 (DOCs p 9 of 24) NUREG Bases markup p B 3.5-8 and B 3.5-16 Retyped ITS Bases p B 3.5-8 and B 3.5-18

SUMMARY OF CHANGES TO ITS SECTION 3.5 - REVISION J

Source of Change	Summary of Change	Affected Pages
Consistency issues	Minor consistency issue corrections have been made. (The word "required" has been added to ITS 3.5.1 Required Actions E.1 and F.1 for consistency with the usage throughout the ITS, since not all ADS valves are required to be Operable; the unit "psig" has been added after the value "1040" in SR 3.5.1.8, since each value should have the unit immediately after it; the SR 3.5.1.4 Bases have been modified to include the information relocated by DOC LA4 (i.e., that the motor-operated cross-tie valves be chain locked closed and the manual cross-tie valve be locked closed); the words "low pressure" have been added to ITS 3.5.2 Condition C and Required Actions A.1 and C.2 to be consistent with a similar addition to Condition A, since the Condition and Required Actions affect only the low pressure ECCS; the word "required" has been added to ITS 3.5.2 Required Action C.2 for consistency with the usage throughout the ITS, since not all low pressure ECCS subsystems are required Operable; and the word "OPERABLE" has been added to the ITS 3.5.2 Bases LCO section, consistent with a similar sentence in ITS 3.4.7 and ITS 3.4.8 Bases.)	**Specification 3.5.1** NUREG ITS markup p 3.5-2 and 3.5-5 NUREG Bases markup p B 3.5-11 Retyped ITS p 3.5-2 and 3.5-5 Retyped ITS Bases p B 3.5-11 **Specification 3.5.2** NUREG ITS markup p 3.5-7 NUREG Bases markup p B 3.5-17 Retyped ITS p 3.5-9 Retyped ITS Bases p B 3.5-20
Editorial	The words "at the 258,000 gallon level" have been changed to "at approximately 258,000 gallons" in the SR 3.5.2.1 and SR 3.5.2.2 Bases.	**Specification 3.5.2** NUREG Bases markup p B 3.5-20 Retyped ITS Bases p B 3.5-24
Technical change	DOC LA2 justified the relocation to the Bases the details in CTS 4.5.D.1.a that the simulated automatic actuation test for the ADS System opens the pilot valves. However, the NUREG and ITS Bases only states that the endpoint of the test is the solenoids, not the pilot valves. The LA DOC has been modified to clearly state that the endpoint is the solenoids and not the pilot valves. This is acceptable since SR 3.5.1.13, which is performed on a similar Frequency, actually opens the ADS valves, including the pilot valves.	**Specification 3.5.1** DOC LA2 (DOCs p 11 of 24)

SUMMARY OF CHANGES TO ITS SECTION 3.5 - REVISION J

Source of Change	Summary of Change	Affected Pages
Technical change	ITS SR 3.5.3.3 requires a stroke test of each RCIC MOV, consistent with CTS 4.5.E.1.c. However, the NUREG does not include a Surveillance Requirement to perform valve stroke testing for the RCIC System. These types of SRs in the CTS are not required to be included in the ITS, and are normally relocated to the IST Program. However, at JAFNPP, these RCIC Valves are not tested per the IST Program. Therefore, JAFNPP will relocate the CTS 4.5.E.1.c RCIC System MOV requirements to the TRM and not include the SR in the ITS; i.e., ITS SR 3.5.3.3 has been deleted and the subsequent SRs have been renumbered. In addition, the check valve testing requirements of CTS 4.5.E.1.e will also be relocated to the TRM, in lieu of relocating this requirement to the IST Program as stated in DOC LA4.	**Specification 3.5.3** CTS markup p 1 of 4 and 2 of 4 DOCs A3, M2, M3, M5, LA1, LA4, L3, and L6 (DOCs p 1 of 7, 2 of 7, 3 of 7, 4 of 7, 5 of 7, 6 of 7, and 7 of 7) NUREG ITS markup p 3.5-12 and 3.5-13 (Insert page 3.5-12 deleted) JFDs CLB2, CLB3, CLB4, DB2, and DB3 (JFDs p 1 of 2) NUREG Bases markup p B 3.5-27, Insert page B 3.5-27, B 3.5-28, and Insert page B 3.5-28 Bases JFDs CLB2, CLB3, CLB4, and DB6 (Bases JFDs p 1 of 3 and 2 of 3) Retyped ITS p 3.5-14 and 3.5-15 Retyped Bases p B 3.5-30, B 3.5-31, and B 3.5-32

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

M8 According to CTS 3.9.F.1, the reactor shall not be made critical unless both LPCI MOV Independent Power Supplies are operable which is effectively MODES 1 and 2. ITS 3.5.1 requires the low pressure core injection subsystems to be Operable in MODES 1, 2 and 3. Since the operability of the LPCI MOV Independent Power Supply effects the OPERABILITY of the associated LPCI subsystem, the operability requirements of LPCI MOV Independent Power Supplies have been extended to MODE 3. This ensures that each LPCI subsystem will remain operable with the required uninterruptable power supply during reactor conditions where there is significant core energy. This change is considered more restrictive and has no adverse effect on safety.

M9 CTS 4.6.E.4 requires the safety/relief valves to be manually opened every 24 months. ITS SR 3.5.1.13 requires this same manual opening but requires the actuation to be initiated on a Staggered Test Basis for each valve solenoid. This will ensure that a different solenoid will be used to actuate the valve every 24 months and is considered more restrictive since the current requirement does not specify which solenoid to use. This change is necessary to ensure both solenoids are tested within any 48 month period.

M10 CTS 3.5.A.5 requires all recirculation pump discharge valves to be Operable prior to reactor startup (or closed if permitted elsewhere in these specifications). ITS 3.5.1 and associated SR 3.5.1.6 also require all recirculation pump discharge valves to be Operable. However, if this requirement can not be met, then ITS SR 3.5.1.6 allows the associated recirculation pump discharge valve to be "de-energized" in the closed position. Requiring the inoperable recirculation pump discharge valve to also be "de-energized" in the closed position represents an additional restriction on plant operation. This change is necessary to ensure the proper flow path for the associated LPCI subsystem.

M11 CTS 4.5.G.3 requires the HPCI System discharge piping to be vented from the high point of the system whenever HPCI is lined up to take suction from the condensate storage tank (CST) on a monthly basis. In ITS SR 3.5.1.1 this requirement must be met whenever HPCI is required to be Operable whether it is aligned to the CST or the suppression pool. This change is considered more restrictive on plant operation but necessary to help prevent a water hammer following an initiation signal.

M12 CTS 3.5.A.1 and 3.5.A.3 require the Core Spray (CS) and Low Pressure Coolant Injection (LPCI) Systems, respectively to be Operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from cold shutdown (this covers MODES 1, 2 in the ITS). CTS 3.5.A

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 (continued)

The TRM will be incorporated by reference into the UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

LA2 The details in CTS 4.5.D.1.a that the simulated automatic actuation test for the Automatic Depressurization System (ADS) opens the pilot valves is proposed to be relocated to the Bases. However, the end point of the test will be the solenoids, not the pilot valves (the JAFNPP S/RVs are two-stage target rock, with a pilot disc and a main disc). The requirement in SR 3.5.1.11 to verify the ADS actuates on an actual (L1) or simulated automatic initiation signal every 24 months, the requirement in SR 3.5.1.13 to open each ADS valve every 24 months (on a STAGGERED TEST BASIS for each valve solenoid), the requirement in LCO 3.5.1 that the ADS function of six safety/relief valves shall be OPERABLE, the definition of OPERABILITY and the applicability of these requirements ensures the appropriate components must be OPERABLE and tested in the required Frequency. As such, these details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications. J 5

LA3 The methods in CTS 4.6.E.4 for verifying the safety/relief valves have opened (i.e., while bypassing steam to the condenser, etc) and the detail that the test must be performed in Run are proposed to be relocated to the Bases. These details are not necessary to ensure Operability of the S/RVs. The requirements of ITS LCO 3.5.1 and the associated SRs are adequate to ensure that ADS is maintained OPERABLE. SR 3.5.1.13 will require each required ADS valve to be manually actuated after reactor steam dome pressure and flow are adequate to perform this test. The Bases for this SR will prescribe the test method and the conditions for performing the test. In addition, the Bases discusses that the pressure and flow conditions will require the plant to be in MODE 1, which has been shown to be an acceptable condition to perform this test. This test will cause a small neutron flux transient which may cause a scram while operating close to the Average Power Range Monitors Neutron Flux-High (Startup) Allowable Value in MODE 2. As such these methods of verification and details that the plant must be in Run are not necessary to be included in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA4 CTS 3.5.A.3.b contains detailed descriptions of the requirements of assuring that the LPCI cross-tie line be isolated. ITS SR 3.5.1.4 requires that the cross tie valves be verified closed and electrical power be removed from the electrically powered motor operator. Additional details on actual valve numbers and method of valve closing presents information that is not required for assuring that the cross-tie be isolated. These additional details are proposed to be relocated to the Bases. As such, these details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LA5 The requirement in CTS 4.6.E.3 concerning the integrity of the nitrogen system and components which provide manual and ADS actuation of the safety/relief valves are proposed to be relocated to the Technical Requirements Manual (TRM). The system will continue to be required to perform its required safety function to be considered OPERABLE. ITS SR 3.5.1.3 is added (refer to M6) to address the important characteristic of whether there is sufficient pneumatic pressure available to permit the actuation of the ADS valves should an accident occur. The Operability requirements of ITS 3.5.1 for ADS valves and the ITS definition of OPERABLE - OPERABILITY are adequate to ensure the ADS valves are maintained capable of performing their specified safety function. In addition, the surveillance being relocated will continue to be performed and will identify degradation of the ADS nitrogen system pressure retention capabilities. As such, this surveillance is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the relocated requirement in the TRM will be controlled by the provisions of 10 CFR 50.59.

LA6 CTS 4.5.G.1, 4.5.G.2, and CTS 4.5.G.3 present the technical detail of the method to be employed to assure that the Core Spray, Low Pressure Coolant Injection and High Pressure Coolant Injection pump discharge lines are full of water (shall be vented from the high point of the system and water flow observed). The detail pertaining to how these Surveillances are to be performed are proposed to be relocated to the Bases. These details are not necessary to ensure the Operability of the ECCS subsystems. The requirements of Specification 3.5.1, ECCS—Operating, and the associated SR 3.5.1.1 are adequate to ensure the ECCS subsystems remain Operable. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the 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 Technical Specifications.

INSERT ACTION A

OR

One low pressure coolant injection (LPCI) pump in both LPCI subsystems inoperable. | 1

ACTIONS (continued)

[L6] [M3] [M3] [M3] [3.5.D.2] PA2 [3.5.C.1.b]

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. HPCI System inoperable. AND ~~One low pressure ECCS injection/spray subsystem is inoperable.~~ Condition A entered. TA2	D.1 Restore HPCI System to OPERABLE status. OR D.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status. (S)	72 hours 72 hours
E. One required ADS valve inoperable. PA1	E.1 Restore ADS valve to OPERABLE status.	14 days
F. One required ADS valve inoperable. AND ~~One low pressure ECCS injection/spray subsystem inoperable.~~ Condition A entered. TA2 PA1	F.1 Restore ADS valve to OPERABLE status. OR F.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status. (S)	72 hours 72 hours
G. Two or more required ADS valves inoperable. OR Required Action and associated Completion Time of Condition C, D, E, or F not met.	G.1 Be in MODE 3. AND G.2 Reduce reactor steam dome pressure to ≤ ~~1~~150~~0~~ psig. DB2	12 hours 36 hours

J J

(continued)

REVISION ~~0~~ J

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.6 [4.5.A.5]	------------------NOTE------------------ Not required to be performed if performed within the previous 31 days. -- Verify each recirculation pump discharge valve ~~[and bypass valve]~~ cycles through one complete cycle of full travel [or is de-energized in the closed position]. DB10 DB10	Once each startup prior to exceeding 25% RTP
SR 3.5.1.7 [4.5.A.1.b] [4.5.A.3]	Verify the following ECCS pumps develop the specified flow rate [against a system head corresponding to the specified reactor pressure] above primary containment pressure DB7 CLB7	~~[In accordance with the Inservice Testing Program or 92 days]~~ CLB2
	SYSTEM / FLOW RATE / NO. OF PUMPS / [SYSTEM HEAD CORRESPONDING TO A REACTOR PRESSURE OF] ABOVE PRIMARY CONTAINMENT PRESSURE DB7	
	Core Spray ≥ ~~[4250]~~ 4265 gpm [1] ≥ [113] psig DB7	
	LPCI ≥ ~~[17,000]~~ 7700 gpm ~~[2]~~ 1 ≥ [20] psig DB7	
SR 3.5.1.8 [4.5.C] [4.5.C.1]	------------------NOTE------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with [reactor pressure] ≤ ~~[1020]~~ 1040 psig and ≥ ~~[920]~~ 970 psig, the HPCI pump can develop a flow rate ≥ ~~[4250]~~ 3400 gpm [against a system head corresponding to reactor pressure]. DB8 DB8 DB7	In accordance with the Inservice Testing Program CLB2 ~~92 days~~

(continued)

REVISION ~~D~~ J

CLB4 INSERT SR 3.5.1.10 NOTE 1

1. For the HPCI System, not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.

CLB5 INSERT SR 3.5.1.12

SR 3.5.1.12	Verify each LPCI motor operated valve independent power supply inverter capacity is adequate to supply and maintain in OPERABLE status the required emergency loads for the design duty cycle.	24 months

recirculation pump suction line

DB1

Division 2

BASES

LOCA due to a

APPLICABLE SAFETY ANALYSES (continued)

e. Adequate long term cooling capability is maintained.

DB1

The limiting single failures are discussed in Reference ~~11~~ 5. For a large ~~discharge~~ pipe break LOCA, failure of the ~~LPCI valve on the unbroken recirculation loop~~ is considered the most severe failure. For a small break LOCA, HPCI failure is the most severe failure. ~~One ADS valve failure is analyzed as a limiting single failure for events requiring ADS operation.~~ The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage.

125 VDC battery

Insert ASA

DB2

10 CFR 50.36(c)(2)(ii) (Ref. 11)

X1

J

The ECCS satisfy Criterion 3 of ~~the NRC Policy Statement~~.

six DB2

LCO

Each ECCS injection/spray subsystem and ~~seven~~ ADS valves are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the two CS subsystems, the two LPCI subsystems, and one HPCI System. The low pressure ECCS injection/spray subsystems are defined as the two CS subsystems and the two LPCI subsystems.

(which includes both pumps per subsystem)

PA1

active component

With less than the required number of ECCS subsystems OPERABLE, the potential exists that during a limiting design basis LOCA concurrent with the worst case single failure, the limits specified in Reference 10 could be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by Reference 10.

PA1

LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below the actual RHR cut in permissive pressure in MODE 3, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling when necessary.

Alignment and operation for decay heat removal includes when the system is being realigned from or to the RHR shutdown cooling mode.

APPLICABILITY

All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3, when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system piping. In MODES 2 and 3, when reactor steam dome pressure

(continued)

BASES

two of the seven DB2

ACTIONS

E.1 (continued)

shows that, assuming a failure of the HPCI System

of an analysis that evaluated the effect of one ADS valve(s) being out of service. ~~Per~~ this analysis, operation of only ~~six~~ five ADS valves will provide the required depressurization. However, overall reliability of the ADS is reduced, because a single active component failure in the OPERABLE ADS valves could result in a reduction in depressurization capability. Therefore, operation with five ADS valves is only allowed for a limited time. The 14 day Completion Time is ~~based on~~ consistent with the recommendations provided in a reliability study cited in Reference 12 and has been found to be acceptable through operating experience.

DB2 PA1 DB5

F.1 and F.2

required PA1

If any one low pressure ECCS required injection/spray subsystem ~~is~~ or one LPCI pump in both LPCI subsystems is inoperable in addition to one inoperable ADS valve, adequate core cooling is ensured by the OPERABILITY of HPCI and the remaining low pressure ECCS injection/spray subsystem(s). However, overall ECCS reliability is reduced because a single active component failure concurrent with a design basis LOCA could result in the minimum required ECCS equipment not being available. Since both a high pressure system (ADS) and a low pressure subsystem are inoperable, a more restrictive Completion Time of 72 hours is required to restore either the low pressure ECCS subsystem(s) or the ADS valve to OPERABLE status. This Completion Time is ~~based on~~ consistent with the recommendations provided in a reliability study cited in Reference 12 and has been found to be acceptable through operating experience.

TA2 DB5 J

G.1 and G.2

required PA5

If any Required Action and associated Completion Time of Condition C, D, E, or F is not met, or if two or more required ADS valves are inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to ≤ 150 psig within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

In addition, plant procedures require the motor operated cross tie valve to be chain-locked closed and the manual cross tie valve to be locked closed.

CLB1

J

DB1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.4 (continued) one or more of motor operated s are

removing the breaker. If the RHR System cross tie valve ~~is~~ open or power has not been removed from the valve ~~operator~~, both LPCI subsystems must be considered inoperable. The 31 day Frequency has been found acceptable, considering that these valves are under strict administrative controls that will ensure the valves continue to remain closed with either control or motive power removed.

CLB6

Cycling open and closed each LPCI motor operated valve independent power supply battery charger AC input breaker and

CLB6

the capability of the supply to become independent from emergency AC power and

SR 3.5.1.5

PA7 DB7 576 624

Verification ~~every 31 days~~ that each LPCI inverter output has a voltage of ≥ [570] V and ≤ [630] V while supplying its respective bus demonstrates that the AC electrical power is available to ensure proper operation of the associated LPCI ~~inboard~~ injection and ~~minimum flow~~ valves and the recirculation pump discharge valve. Each inverter must be OPERABLE for the associated LPCI subsystem to be OPERABLE. The 31 day Frequency has been found acceptable based on ~~engineering judgment and~~ operating experience.

heat exchanger bypass

DB1

PA7

and battery charger AC input breaker

CLB6

SR 3.5.1.6

DB1

Cycling the recirculation pump discharge ~~[and bypass]~~ valves through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will close when required. Upon initiation of an automatic LPCI subsystem injection signal, these valves are required to ~~be~~ close~~d~~ to ensure full LPCI subsystem flow injection in the reactor via the recirculation jet pumps. De-energizing the valve in the closed position will also ensure the proper flow path for the LPCI subsystem. Acceptable methods of de-energizing the valve include de-energizing breaker control power, racking out the breaker or removing the breaker.

PA1

The specified Frequency is once during reactor startup before THERMAL POWER is > 25% RTP. However, this SR is modified by a Note that states the Surveillance is only required to be performed if the last performance was more than 31 days ago. ~~Therefore, implementation of this Note requires this test to be performed during reactor startup before exceeding 25% RTP.~~ Verification during reactor

PA4

(continued)

BASES

PA3

DB8 brackets

REFERENCES (continued)

3. UFSAR, Section ~~[6.3.2.2.1]~~. 6.4.1
4. UFSAR, Section ~~[6.3.2.2.2]~~. 6.4.2
5. ~~FSAR, Section [15.2.8]~~. NEDC-31317P, Revision 2, James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA, Loss-of-Coolant Accident Analysis, April 1993.
6. UFSAR, Section ~~[15.6.4]~~. 14.6.1.5
7. UFSAR, Section ~~[15.6.5]~~ 14.6.1.3
8. 10 CFR 50, Appendix K.
9. UFSAR, Section ~~[6.3.3]~~. 6.5
10. 10 CFR 50.46.
11. ~~FSAR, Section [7.3.1.2.2]~~. 10 CFR 50.36(c)(2)(ii) X1 1J
12. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975. PA6
13. UFSAR, Section ~~[6.3.3.3]~~ 4.4.5

J

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. HPCI System inoperable.	C.1	Verify by administrative means RCIC System is OPERABLE.	Immediately
	AND		
	C.2	Restore HPCI System to OPERABLE status.	14 days
D. HPCI System inoperable. AND Condition A entered.	D.1	Restore HPCI System to OPERABLE status.	72 hours
	OR		
	D.2	Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	72 hours
E. One required ADS valve inoperable.	E.1	Restore required ADS valve to OPERABLE status.	14 days
F. One required ADS valve inoperable. AND Condition A entered.	F.1	Restore required ADS valve to OPERABLE status.	72 hours
	OR		
	F.2	Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	72 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.6	-----NOTE----- Not required to be performed if performed within the previous 31 days. ---------- Verify each recirculation pump discharge valve cycles through one complete cycle of full travel or is de-energized in the closed position.	Once each startup prior to exceeding 25% RTP
SR 3.5.1.7	Verify the following ECCS pumps develop the specified flow rate against a system head corresponding to the specified reactor pressure above primary containment pressure. SYSTEM / FLOW RATE / NO. OF PUMPS / SYSTEM HEAD CORRESPONDING TO A REACTOR PRESSURE ABOVE PRIMARY CONTAINMENT PRESSURE OF Core Spray / ≥ 4265 gpm / 1 / ≥ 113 psi LPCI / ≥ 7700 gpm / 1 / ≥ 20 psi	In accordance with the Inservice Testing Program
SR 3.5.1.8	-----NOTE----- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ---------- Verify, with reactor pressure ≤ 1040 psig and ≥ 970 psig, the HPCI pump can develop a flow rate ≥ 3400 gpm against a system head corresponding to reactor pressure.	In accordance with the Inservice Testing Program J

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.5.1.9	------------------NOTE------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 165 psig, the HPCI pump can develop a flow rate ≥ 3400 gpm against a system head corresponding to reactor pressure.	24 months
SR 3.5.1.10	------------------NOTES------------------ 1. For the HPCI System, not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. 2. Vessel injection/spray may be excluded. -- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.1.11	------------------NOTE------------------ Valve actuation may be excluded. -- Verify the ADS actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.1.12	Verify each LPCI motor operated valve independent power supply inverter capacity is adequate to supply and maintain in OPERABLE status the required emergency loads for the design duty cycle.	24 months

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. The core is maintained in a coolable geometry; and

e. Adequate long term cooling capability is maintained.

The limiting single failures are discussed in Reference 5. For a LOCA due to a large recirculation pump suction line pipe break, failure of the Division 2 125 VDC battery is considered the most severe failure. For a small break LOCA, HPCI failure is the most severe failure. In the analysis of events requiring ADS operation, it is assumed that only five of the seven ADS valves operate. Since six ADS valves are required to be OPERABLE, the explicit assumption of the failure of an ADS valve is not considered in the analysis. The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage.

The ECCS satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 11).

LCO

Each ECCS injection/spray subsystem and six ADS valves are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the two CS subsystems, the two LPCI subsystems (which includes both pumps per subsystem), and one HPCI System. The low pressure ECCS injection/spray subsystems are defined as the two CS subsystems and the two LPCI subsystems.

With less than the required number of ECCS subsystems OPERABLE, the potential exists that during a limiting design basis LOCA concurrent with the worst case single active component failure, the limits specified in Reference 10 could be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by Reference 10.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the inoperable low pressure ECCS subsystem(s) cannot be restored to OPERABLE status within the associated Completion Time, 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 and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

If the HPCI System is inoperable and the RCIC System is verified to be OPERABLE, the HPCI System must be restored to OPERABLE status within 14 days. In this condition, adequate core cooling is ensured by the OPERABILITY of the redundant and diverse low pressure ECCS injection/spray subsystems in conjunction with ADS. Also, the RCIC System will automatically provide makeup water at most reactor operating pressures. Verification of RCIC OPERABILITY immediately is therefore required when HPCI is inoperable. This may be performed as an administrative check by examining logs or other information to determine if RCIC is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of the RCIC System. If the OPERABILITY of the RCIC System cannot be verified, however, Condition G must be immediately entered. If a single active component fails concurrent with a design basis LOCA, there is a potential, depending on the specific failure, that the minimum required ECCS equipment will not be available. A 14 day Completion Time is consistent with the recommendations provided in a reliability study cited in Reference 12 and has been found to be acceptable through operating experience.

D.1 and D.2

If any one low pressure ECCS injection/spray subsystem or one LPCI pump in both LPCI subsystems is inoperable in addition to an inoperable HPCI System, the inoperable low pressure ECCS injection/spray subsystem(s) or the HPCI System must be restored to OPERABLE status within 72 hours.

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

In this condition, adequate core cooling is ensured by the OPERABILITY of the ADS and the remaining low pressure ECCS subsystems. However, the overall ECCS reliability is significantly reduced because a single active component failure in one of the remaining OPERABLE subsystems concurrent with a design basis LOCA may result in the ECCS not being able to perform its intended safety function. Since both a high pressure system (HPCI) and low pressure subsystem are inoperable, a more restrictive Completion Time of 72 hours is required to restore either the HPCI System or the low pressure ECCS injection/spray subsystem to OPERABLE status. This Completion Time is consistent with the recommendations provided in a reliability study cited in Reference 12 and has been found to be acceptable through operating experience.

E.1

The LCO requires six ADS valves to be OPERABLE in order to provide the ADS function. Reference 5 contains the results of an analysis that evaluated the effect of two of the seven ADS valves being out of service. This analysis shows that, assuming a failure of the HPCI System, operation of only five ADS valves will provide the required depressurization. However, overall reliability of the ADS is reduced, because a single active component failure in the OPERABLE ADS valves could result in a reduction in depressurization capability. Therefore, operation with five ADS valves is only allowed for a limited time. The 14 day Completion Time is consistent with the recommendations provided in a reliability study cited in Reference 12 and has been found to be acceptable through operating experience.

F.1 and F.2

If any one low pressure ECCS injection/spray subsystem or one LPCI pump in both LPCI subsystems is inoperable in addition to one required ADS valve inoperable, adequate core cooling is ensured by the OPERABILITY of HPCI and the remaining low pressure ECCS injection/spray subsystem(s). However, overall ECCS reliability is reduced because a

(continued)

BASES

ACTIONS

F.1 and F.2 (continued)

single active component failure concurrent with a design basis LOCA could result in the minimum required ECCS equipment not being available. Since both a high pressure system (ADS) and low pressure subsystem(s) are inoperable, a more restrictive Completion Time of 72 hours is required to restore either the low pressure ECCS subsystem(s) or the ADS valve to OPERABLE status. This Completion Time is consistent with the recommendations provided in a reliability study cited in Reference 12 and has been found to be acceptable through operating experience.

G.1 and G.2

If any Required Action and associated Completion Time of Condition C, D, E, or F is not met, or if two or more required ADS valves are inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to ≤ 150 psig within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

H.1

When multiple ECCS subsystems are inoperable, as stated in Condition H, the plant is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

The flow path piping has the potential to develop voids and pockets of entrained air. Maintaining the pump discharge lines of the HPCI System, CS System, and LPCI subsystems full of water ensures that the ECCS will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent a water hammer following an ECCS

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.2 (continued)

includes when the system is being realigned from or to the RHR shutdown cooling mode. At the low pressures and decay heat loads associated with operation in MODE 3 with reactor steam dome pressure less than the shutdown cooling permissive pressure, a reduced complement of low pressure ECCS subsystems should provide the required cooling, thereby allowing operation of RHR shutdown cooling, when necessary.

SR 3.5.1.3

Verification every 31 days that ADS pneumatic supply header pressure is ≥ 95 psig ensures adequate pneumatic pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The design pneumatic supply pressure requirements for the accumulator are such that, following a failure of the pneumatic supply to the accumulator, at least one valve actuation can occur with the drywell at 70% of design pressure (Ref. 13). The ECCS safety analysis assumes only one actuation to achieve the depressurization required for operation of the low pressure ECCS. This minimum required pressure of ≥ 95 psig is provided by the ADS nitrogen supply. The 31 day Frequency takes into consideration administrative controls over operation of the pneumatic system and alarms for low pneumatic pressure.

SR 3.5.1.4

Verification every 31 days that the RHR System cross tie valves are closed and power to the motor operated valve is disconnected ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other LPCI subsystem. Acceptable methods of removing power to the operator include de-energizing breaker control power or racking out or removing the breaker. If one or more of the RHR System cross tie valves are open or power has not been removed from the motor operated valve, both LPCI subsystems must be considered inoperable. In addition, plant procedures require the motor operated cross tie valve to be chain-locked closed and the manual cross tie valve to be locked closed. The 31 day Frequency has been found acceptable,

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.4 (continued)

considering that these valves are under strict administrative controls that will ensure the valves continue to remain closed with either control or motive power removed.

SR 3.5.1.5

Cycling open and closed each LPCI motor operated valve independent power supply battery charger AC input breaker and verification that each LPCI inverter output has a voltage of ≥ 576 V and ≤ 624 V while supplying its respective bus demonstrates the capability of the supply to become independent from emergency AC power and that the AC electrical power is available to ensure proper operation of the associated LPCI injection and heat exchanger bypass valves and the recirculation pump discharge valve. Each inverter and battery charger AC input breaker must be OPERABLE for the associated LPCI subsystem to be OPERABLE. The 31 day Frequency has been found acceptable based on operating experience.

SR 3.5.1.6

Cycling the recirculation pump discharge valves through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will close when required. Upon initiation of an automatic LPCI subsystem injection signal, these valves are required to close to ensure full LPCI subsystem flow injection in the reactor via the recirculation jet pumps. De-energizing the valve in the closed position will also ensure the proper flow path for the LPCI subsystem. Acceptable methods of de-energizing the valve include de-energizing breaker control power, racking out the breaker or removing the breaker.

The specified Frequency is once during reactor startup before THERMAL POWER is > 25% RTP. However, this SR is modified by a Note that states the Surveillance is only required to be performed if the last performance was more than 31 days ago. Verification during reactor startup prior to reaching > 25% RTP is an exception to the normal Inservice Testing Program generic valve cycling Frequency of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.6 (continued)

92 days, but is considered acceptable due to the demonstrated reliability of these valves. If the valve is inoperable and in the open position, the associated LPCI subsystem must be declared inoperable.

SR 3.5.1.7, SR 3.5.1.8, and SR 3.5.1.9

The performance requirements of the low pressure ECCS pumps are determined through application of the 10 CFR 50, Appendix K criteria (Ref. 8). This periodic Surveillance is performed (in accordance with the ASME Code, Section XI, requirements for the ECCS pumps) to verify that the ECCS pumps will develop at least the flow rates required by the respective analyses. The low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of Reference 10. The pump flow rates are verified against a system head equivalent to the RPV pressure expected during a LOCA. The total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during a LOCA. These values may be established during preoperational testing.

The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow against a system head corresponding to reactor pressure is tested at both the higher and lower operating ranges of the system. The required system head should overcome the RPV pressure and associated discharge line losses. Adequate reactor steam pressure must be available to perform these tests. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the HPCI System diverts steam flow. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these tests. Adequate reactor steam pressure must be ≥ 970 psig to perform SR 3.5.1.8 and > 150 psig to perform SR 3.5.1.9. Adequate steam flow is represented by at least one turbine bypass valve open or main turbine generator load is greater than 100 MWe. Reactor startup is allowed prior to performing the low pressure Surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance test is short. The

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.7, SR 3.5.1.8, and SR 3.5.1.9 (continued)

reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure test has been satisfactorily completed and there is no indication or reason to believe that HPCI is inoperable. Therefore, SR 3.5.1.8 and SR 3.5.1.9 are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for performing the flow test after the required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides reasonable time to complete the SRs.

The Frequency for SR 3.5.1.7 and SR 3.5.1.8 is in accordance with the Inservice Testing Program requirements. The 24 month Frequency for SR 3.5.1.9 is based on the need to perform the Surveillance under the conditions that apply during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.1.10

The ECCS subsystems are required to actuate automatically to perform their design functions. This Surveillance verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of HPCI, CS, and LPCI will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup and actuation of all automatic valves to their required positions. The HPCI System actual or simulated automatic actuation test must be performed with adequate steam pressure for verification of automatic pump startup. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the HPCI System diverts steam flow. Thus, sufficient time is allowed after adequate pressure and flow are achieved to perform this test associated with the HPCI System. Adequate reactor steam dome pressure is > 150 psig. Adequate steam flow is represented by at least one turbine bypass valve open. This SR also ensures that the HPCI System will automatically

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.10 (continued)

restart on an RPV low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) trip. In addition, this SR also ensures that the HPCI suction is automatically transferred from the CSTs to the suppression pool on high suppression pool water level or low CST water level. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlaps this Surveillance to provide complete testing of the assumed safety function.

For CS and LPCI, the 24 month Frequency is based on the need to perform the 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. For HPCI, the 24 month Frequency is based on the need to perform the surveillance under conditions that apply during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by two Notes. Note 1 states that for the HPCI System, the Surveillance is not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for performing the actual or simulated automatic actuation for the HPCI System after the required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides reasonable time to complete the SR. Note 2 excludes vessel injection/spray during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

SR 3.5.1.11

The ADS designated S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to demonstrate that the mechanical portions of the ADS function (i.e., solenoids) operate as designed when initiated either by an actual or simulated initiation signal, causing proper actuation of all the required components. SR 3.5.1.13 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.11 (continued)

overlap this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform the 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 SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes valve actuation since the valves are individually tested in accordance with SR 3.5.1.13. This prevents the possibility of an RPV pressure blowdown.

SR 3.5.1.12

A LPCI motor operated valve independent power supply subsystem inverter test is a test of the inverter's capability, as found, to satisfy the design requirements (inverter duty cycle). The discharge rate and test length correspond to the design duty cycle requirements. 5

The Frequency of 24 months is acceptable, given plant conditions required to perform the test and the other requirements existing to ensure adequate LPCI inverter performance during the 24 month interval. In addition, the Frequency is intended to be consistent with expected fuel cycle lengths.

SR 3.5.1.13

A manual actuation of each required ADS valve is performed while bypassing main steam flow to the condenser and observing ≥ 10% closure of the turbine bypass valves to verify that the valve and solenoid are functioning properly and that no blockage exists in the S/RV discharge lines. This can also be demonstrated by the response of the turbine control or bypass valve or by a change in the measured flow or by any other method suitable to verify steam flow.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.13 (continued)

Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the ADS valves divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this SR. Adequate pressure at which this SR is to be performed is ≥ 970 psig (the pressure consistent with vendor recommendations). Adequate steam flow is represented by at least two or more turbine bypass valves open or total steam flow ≥ 10^6 lb/hr. These conditions will require the plant to be in MODE 1, which has been shown to be an acceptable condition to perform this test. This test causes a small neutron flux transient which may cause a scram in MODE 2 while operating close to the Average Power Range Monitors Neutron Flux – High (Startup) Allowable Value. Reactor startup is allowed prior to performing this SR because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure and flow are reached is sufficient to achieve stable conditions and provides adequate time to complete the Surveillance. SR 3.5.1.11 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function.

The Frequency of 24 months on a STAGGERED TEST BASIS ensures that both solenoids for each ADS valve are alternately tested. The Frequency is based on the need to perform the Surveillance under the conditions that apply during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 6.4.3.

(continued)

BASES

REFERENCES (continued)

2. UFSAR, Section 6.4.4.

3. UFSAR, Section 6.4.1.

4. UFSAR, Section 6.4.2.

5. NEDC-31317P, Revision 2, James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA, Loss-of-Coolant Accident Analysis, April 1993.

6. UFSAR, Section 14.6.1.5.

7. UFSAR, Section 14.6.1.3.

8. 10 CFR 50, Appendix K.

9. UFSAR, Section 6.5.

10. 10 CFR 50.46.

11. 10 CFR 50.36(c)(2)(ii).

12. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), Recommended Interim Revisions to LCOs for ECCS Components, December 1, 1975.

13. UFSAR, Section 4.4.5.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.2 ECCS—Shutdown

LCO 3.5.2 Two low pressure ECCS injection/spray subsystems shall be OPERABLE.

[3.5.F.1]
[3.5.F.2]
[3.5.F.3]

X1

APPLICABILITY: MODE 4,
MODE 5, except with the spent fuel storage pool gates removed and water level ≥ [23 ft] 22 ft 2 inches over the top of the reactor pressure vessel flange.

[3.5.F.1]
[3.5.F.2]
[3.5.F.3][M3]

ACTIONS

low pressure PA2

CONDITION	REQUIRED ACTION		COMPLETION TIME
[L1] A. One required low pressure ECCS injection/spray subsystem inoperable.	A.1	Restore required low pressure ECCS injection/spray subsystem to OPERABLE status.	4 hours
[3.5.F.4] B. Required Action and associated Completion Time of Condition A not met.	B.1	Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
[3.5.F.4] C. Two required low pressure ECCS injection/spray subsystems inoperable.	C.1	Initiate action to suspend OPDRVs.	Immediately
	AND		
	PA2 C.2	Restore one required low pressure ECCS injection/spray subsystem to OPERABLE status.	4 hours

(continued)

Typ all pages

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.2 ECCS—Shutdown

BASES

BACKGROUND

A description of the Core Spray (CS) System and the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System is provided in the Bases for LCO 3.5.1, "ECCS—Operating."

APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated loss of coolant accident (LOCA). The long term cooling analysis following a design basis LOCA (Ref. 1) demonstrates that only one low pressure ECCS injection/spray subsystem is required, post LOCA, to maintain adequate reactor vessel water level in the event of an inadvertent vessel draindown. It is reasonable to assume, based on engineering judgement, that while in MODES 4 and 5, one low pressure ECCS injection/spray subsystem can maintain adequate reactor vessel water level. To provide redundancy, a minimum of two low pressure ECCS injection/spray subsystems are required to be OPERABLE in MODES 4 and 5.

The low pressure ECCS subsystems satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36(c)(2)(ii) (Ref. 2) X1

LCO DB1

Two low pressure ECCS injection/spray subsystems are required to be OPERABLE. The low pressure ECCS injection/spray subsystems consist of two CS subsystems and two LPCI subsystems. Each CS subsystem consists of one motor driven pump(s), piping, and valves to transfer water from both the suppression pool or condensate storage tank(s) (CST(s)) to the reactor pressure vessel (RPV). The CST suction source consists of two CSTs connected in parallel. DB1 Each OPERABLE LPCI subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool to the RPV. Only a single LPCI pump is required per subsystem because of the larger injection capacity in relation to a CS subsystem. In MODES 4 and 5, the RHR System cross tie valve(s) ~~is~~ are not required to be closed. DB2 J PA

(continued)

BASES

during alignment and operation for decay heat removal

PA1

LCO (continued)

One LPCI subsystem may be aligned for decay heat removal and considered OPERABLE ~~for the ECCS function~~, if ~~it can be~~ capable of being manually realigned (remote or local) to the LPCI mode and is not otherwise inoperable. Because of low pressure and low temperature conditions in MODES 4 and 5, sufficient time will be available to manually align and initiate LPCI subsystem operation to provide core cooling prior to postulated fuel uncovery.

PA1

PA1

Alignment and operation for decay heat removal includes when the system is realigned from or to the RHR shutdown cooling mode.

J

APPLICABILITY

OPERABILITY of the low pressure ECCS injection/spray subsystems is required in MODES 4 and 5 to ensure adequate coolant inventory and sufficient heat removal capability for the irradiated fuel in the core in case of an inadvertent draindown of the vessel. Requirements for ECCS OPERABILITY during MODES 1, 2, and 3 are discussed in the Applicability section of the Bases for LCO 3.5.1. ECCS subsystems are not required to be OPERABLE during MODE 5 with the spent fuel storage pool gates removed and the water level maintained at ≥ ~~23 ft~~ 22 ft 2 inches above the RPV flange. This provides sufficient coolant inventory to allow operator action to terminate the inventory loss prior to fuel uncovery in case of an inadvertent draindown.

X2

The Automatic Depressurization System is not required to be OPERABLE during MODES 4 and 5 because the RPV pressure is ≤ 150 psig, and the CS System and the LPCI subsystems can provide core cooling without any depressurization of the primary system.

The High Pressure Coolant Injection System is not required to be OPERABLE during MODES 4 and 5 since the low pressure ECCS injection/spray subsystems can provide sufficient flow to the vessel.

ACTIONS

A.1 and B.1

If any one required low pressure ECCS injection/spray subsystem is inoperable, the inoperable subsystem ~~must~~ be restored to OPERABLE status in 4 hours. In this ~~C~~ondition, the remaining OPERABLE subsystem can provide sufficient vessel flooding capability to recover from an inadvertent vessel draindown. However, overall system reliability is

PA1

(continued)

Revision J

BASES

ACTIONS

C.1, C.2, D.1, D.2, and D.3 (continued)

necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, the Surveillance may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

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The 4 hour Completion Time to restore at least one low pressure ECCS injection/spray subsystem to OPERABLE status ensures that prompt action will be taken to provide the required cooling capacity or to initiate actions to place the plant in a condition that minimizes any potential fission product release to the environment.

PA1

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1 and SR 3.5.2.2

The minimum water level of [~~12 ft 2 inches~~] 10.33 ft required for the suppression pool is periodically verified to ensure that the suppression pool will provide adequate net positive suction head (NPSH) for the CS System and LPCI subsystem pumps, recirculation volume, and vortex prevention. With the suppression pool water level less than the required limit, all ECCS injection/spray subsystems are inoperable unless they are aligned to an OPERABLE CST.

DB2 unless otherwise noted

When suppression pool level is < [~~12 ft 2 inches~~] 10.33 ft, the CS System is considered OPERABLE only if it can take suction from both ~~the~~ CSTs, and the CST water level is sufficient to provide the required NPSH for the CS pump. Therefore, a verification that either the suppression pool water level is ≥ [~~12 ft 2 inches~~] 10.33 ft or that CS is aligned to take suction from both ~~the~~ CSTs and the CSTs contain ≥ [~~150,000~~] 354,000 gallons (two tanks) of water, equivalent to ~~[12 ft]~~ 324 inches (27 ft), ensures that the CS System can supply at least [50,000] 258,000 gallons of makeup water to the RPV. The CS suction is uncovered at ~~the~~ [100,000] gallons levels (two tanks) approximately. However, as noted, only one required CS subsystem may take credit for the CST option during OPDRVs. During OPDRVs, the volume in the CSTs may not provide adequate makeup if the RPV were completely drained. Therefore, only one CS subsystem is allowed to use the CSTs. This ensures the other required ECCS subsystem has adequate makeup volume.

An excess amount of water remains as a supplementary volume and to ensure adequate CS pump NPSH.

J

J

(continued)

Revision J

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.4 (continued) Capable of being

during alignment and operation for shutdown cooling

~~for the ECCS function~~ if ~~all the required valves in the LPCI flow path can be~~ manually realigned (remote or local) to the LPCI mode and ~~allow injection into the RPV, and the system is~~ not otherwise inoperable. This will ensure adequate core cooling if an inadvertent RPV draindown should occur.

PA1

REFERENCES

1. ~~UF~~SAR, Section ~~[6.3.2]~~ 6.5.3. PA2 DB3
2. 10 CFR 50.36(c)(2)(ii). X1

PA1

Alignment and operation for decay heat removal includes when the system is being realigned from or to the RHR shutdown cooling mode. Because of the low pressure and low temperature conditions in MODE 4 and 5 sufficient time will be available to manually align and initiate LPCI subsystem operation to provide core cooling prior to postulated fuel uncovery.

PA1

1/J

Revision J

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.2 ECCS – Shutdown

LCO 3.5.2 Two low pressure ECCS injection/spray subsystems shall be OPERABLE.

APPLICABILITY: MODE 4,
MODE 5, except with the spent fuel storage pool gates removed and water level ≥ 22 ft 2 inches over the top of the reactor pressure vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore required low pressure ECCS injection/spray subsystem to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
C. Two required low pressure ECCS injection/spray subsystems inoperable.	C.1 Initiate action to suspend OPDRVs. AND C.2 Restore one required low pressure ECCS injection/spray subsystem to OPERABLE status.	Immediately 4 hours

(continued)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.2 ECCS – Shutdown

BASES

BACKGROUND

A description of the Core Spray (CS) System and the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System is provided in the Bases for LCO 3.5.1, "ECCS – Operating."

APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated loss of coolant accident (LOCA). The long term cooling analysis following a design basis LOCA (Ref. 1) demonstrates that only one low pressure ECCS injection/spray subsystem is required, post LOCA, to maintain adequate reactor vessel water level in the event of an inadvertent vessel draindown. It is reasonable to assume, based on engineering judgement, that while in MODES 4 and 5, one low pressure ECCS injection/spray subsystem can maintain adequate reactor vessel water level. To provide redundancy, a minimum of two low pressure ECCS injection/spray subsystems are required to be OPERABLE in MODES 4 and 5.

The low pressure ECCS subsystems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

LCO

Two low pressure ECCS injection/spray subsystems are required to be OPERABLE. The low pressure ECCS injection/spray subsystems consist of two CS subsystems and two LPCI subsystems. Each CS subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool or from both condensate storage tanks (CSTs) to the reactor pressure vessel (RPV). The CST suction source consists of two CSTs connected in parallel. Each OPERABLE LPCI subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool to the RPV. Only a single LPCI pump is required per subsystem because of the larger injection capacity in relation to a CS subsystem. In MODES 4 and 5, the RHR System cross tie valves are not required to be closed.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1 and SR 3.5.2.2 (continued)

equivalent to 324 inches (27 ft), ensures that the CS System can supply at least 50,000 gallons of makeup water to the RPV. An excess amount of water remains as a supplementary volume and to ensure adequate CS pump NPSH. The CS suction is uncovered at approximately 258,000 gallons (two tanks). However, as noted, only one required CS subsystem may take credit for the CST option during OPDRVs. During OPDRVs, the volume in the CSTs may not provide adequate makeup if the RPV were completely drained. Therefore, only one CS subsystem is allowed to use the CSTs. This ensures the other required ECCS subsystem has adequate makeup volume. 15

The 12 hour Frequency of these SRs was developed considering operating experience related to suppression pool water level and CST water level variations and instrument drift during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool or CST water level condition.

SR 3.5.2.3, SR 3.5.2.5, and SR 3.5.2.6

The Bases provided for SR 3.5.1.1, SR 3.5.1.7, and SR 3.5.1.10 are applicable to SR 3.5.2.3, SR 3.5.2.5, and SR 3.5.2.6, respectively.

SR 3.5.2.4

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of

(continued)

Specification 3.5.3

A1

JAFNPP

~~3.5 (Cont'd)~~

[3.5.3] ~~E.~~ Reactor Core Isolation Cooling (RCIC) System

[LCO 3.5.3] ~~1.~~ The RCIC System shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 150 psig and reactor coolant temperature is greater than 212°F except from the

[Applicability]

[ACTION A] time that the RCIC System is made or found to be inoperable for any reason, continued reactor power operation is permissible during the succeeding 7 days unless the system is made operable earlier provided that during these 7 days the HPCI System is operable. 14 L1

M1 MODE 3 in 12 hours

[ACTION B] ~~2.~~ If the requirements of 3.5.E cannot be met, the reactor shall be placed in the cold condition and pressure ~~less~~ ~~than~~ ≤ 150 psig within 24 hours. 36 L2

L5

~~3.~~ ~~Low power physics testing and reactor operator~~ training shall be permitted with inoperable components as specified in 3.5.E.2 above, provided that reactor coolant temperature is ≤212°F.

A5

~~4.5 (Cont'd)~~

[3.5.3] ~~E.~~ Reactor Core Isolation Cooling (RCIC) System

[SR 3.5.3.3 NOTE] [SR 3.5.3.4 NOTE] [SR 3.5.3.5 NOTE 1]

1\. RCIC System testing shall be performed as follows provided a reactor steam supply is available. If steam is not available at the time the surveillance test is scheduled to be performed, the test shall be performed within ten days of continuous operation from the time steam becomes available. M2 12 hours

~~Item~~ Surveillance	Frequency
[SR 3.5.3.5] a. actual or L3 ~~Simulated Automatic~~ Actuation (and Restart*) Test LA1	Once per 24 Months <ADD SR 3.5.3.5 NOTE 2> A3
[SR 3.5.3.2] b. Verify that each valve (manual, power operated or automatic) in the system flowpath that is not locked, sealed or otherwise secured in position, is in the correct position.	Once per 31 Days
~~c. Motor Operated Valve Operability~~	~~Once per 92 Days~~ LA4

~~* Automatic restart on a low water level signal which is subsequent to a high water level trip.~~ LA1

A1

~~3.5 (cont'd)~~

~~4.5 (cont'd)~~

[SR 3.5.3.3]

~~Item~~ SURVEILLANCE	Frequency
d. Flow Rate Test - The RCIC pump shall deliver at least 400 gpm against a system head corresponding to a reactor vessel pressure of 1195 psig to ~~150~~ psig.	Once per 92 Days
~~e. Testable Check Valves~~	~~Tested for operability any time the reactor is in the cold condition exceeding 48 hours, if operability tests have not been performed during the preceding 92 days.~~
f. Logic System Functional Test	Once per 24 Months

L6

M3

with reactor pressure ≥ 970 psig and ≤ 1040 psig [SR 3.5.3.3]

with reactor pressure ≤ 165 psig, every 24 months [SR 3.5.3.4]

LA4

See ITS: 3.3.5.2

[Required Action A.1]

2. When it is determined that the RCIC System is inoperable at a time when it is required to be operable, the HPCI System shall be verified to be operable immediately ~~and daily thereafter~~.

L4

DISCUSSION OF CHANGES
ITS: 3.5.3 - RCIC SYSTEM

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 CTS 3.5.G.1 requires the RCIC pump to be considered inoperable when the associated pump discharge piping cannot be maintained in a filled condition. This will require entry into CTS 3.5.E where 7 days (L1) is allowed to restore the RCIC System to Operable status. In the ITS, the requirement that the RCIC discharge piping must be filled is reflected in SR 3.5.3.1. Therefore, since this SR is directly related to the operability requirements of the RCIC System, this cross reference can be deleted and this change considered administrative. This change is consistent with NUREG-1433, Revision 1.

A3 CTS 4.5.E.1.a (ITS SR 3.5.3.5) is modified by Note 2 that excludes vessel injection during the Surveillance. The Bases indicates that this test must include actuation of all automatic valves to their required positions. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance. This Note, therefore, is explicit recognition that ITS SR 3.5.3.5 can be satisfied by a series of overlapping tests. Since surveillance testing of RCIC (CTS 4.5.E.1.a) does not presently require actual injection, and is currently satisfied by a series of overlapping tests, the addition of the Note excluding vessel injection is an administrative change.

A4 Not used.

A5 CTS 3.5.E.3 does not require the Reactor Core Isolation Cooling (RCIC) System to be Operable during low power physics testing and during reactor operator training provided the reactor coolant temperature is ≤ 212°F. This explicit requirement is not retained in the ITS. CTS 3.5.E.1 does not require the RCIC System to be Operable when the reactor coolant temperature is ≤ 212°F. Therefore, since there are no Operability requirements for the RCIC System during the conditions of CTS 3.5.E.3, the allowances provided are meaningless and therefore this deletion is considered administrative. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.5.E.2 requires the reactor to be placed in the cold condition and pressure less than 150 psig within 24 hours when CTS 3.5.E cannot be met. This requirement is proposed to be replaced by ITS 3.5.3 Required Actions B.1 and B.2 which require the plant be in MODE 3 within 12 hours and to reduce reactor steam dome pressure to ≤ 150 psig within 36 hours (see L2) under the same condition. Based on operating experience, this Completion Time limit still allows for an orderly transition to MODE 3 without challenging plant systems. This change is more restrictive because it provides an additional requirement to place the plant in MODE 3 in 12 hours prior to requiring reactor steam dome pressure to be ≤ 150 psig.

M2 CTS 4.5.E.1 requirement, to permit up to 10 days of continuous operation from the time steam becomes available until RCIC Surveillances need to be performed, is being changed. The Note to ITS SR 3.5.3.3 and SR 3.5.3.4 and Note 1 of SR 3.5.3.5 allow only 12 hours from the time reactor steam pressure and flow are adequate to perform the test. The 12 hours allows sufficient time to achieve stable conditions for testing and provides a reasonable time to complete the SR without impacting plant operation. Reducing the allowable time to perform the test, from 10 days to 12 hours, imposes additional operational limitations. This change will require that the actual surveillances be performed sooner in the plant startup, and thereby demonstrate RCIC Operability sooner than current requirements dictate. Therefore, this change is considered more restrictive but necessary to ensure Operability within a reasonable time period when the equipment is required to be Operable.

M3 The CTS 4.5.E.1.d requirement, that RCIC deliver at least 400 gpm against a system head corresponding to a reactor vessel pressure of 1,195 psig to 150 psig, is being divided into two separate Surveillance Requirements SR 3.5.3.3 and SR 3.5.3.4. ITS SR 3.5.3.3, will require demonstration of the RCIC pump capability at the high reactor vessel pressure each 92 days, with reactor pressure ≥ 970 psig and ≤ 1040 psig. Reactor pressures of ≥ 970 psig and ≤ 1040 psig represents a nominal value at rated conditions within the CTS required band for testing. This pressure range represents conditions of lower driving pressure for the RCIC turbine and thus, a more restrictive condition under which to provide the required flow. ITS SR 3.5.3.4 will require demonstration of the RCIC pump capability at the low reactor vessel pressure every 24 months with reactor pressure ≤ 165 psig. Reactor pressure of ≤ 165 psig is near the lower limit (i.e., ≥ 150 psig) of operability/capability of the RCIC turbine, yet provides a 15 psig range above the lower limit in which to conduct the test. CTS required that the RCIC test confirm the capability of the pump at 150 psig. As a practical consideration, the test is performed when sufficient pressure is available at near 150 psig. To require the test at ≤ 150 psig would be to require a test

TECHNICAL CHANGES - MORE RESTRICTIVE

M3 (continued)

of the capability of the pump outside the required operability range. This change will ensure the RCIC System is tested at both the high and low pressures at the proposed Frequencies and is therefore considered more restrictive on plant operation but necessary to ensure RCIC remains Operable over its full operating range.

M4 CTS 4.5.G.3 requires the RCIC System discharge piping to be vented from the high point of the system whenever RCIC is lined up to take suction from the condensate storage tank (CST). In ITS SR 3.5.3.1, this requirement must be met whenever RCIC is required to be Operable, not just when RCIC is lined up to take suction from the CST. This change is considered more restrictive on plant operation but necessary to help prevent a water hammer following an initiation signal.

M5 Not used. J

M6 Not used.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The details of CTS 4.5.E.1.a footnote *, that states "automatic restart on a low water level signal which is subsequent to a high water level signal", are proposed to be relocated to the Bases. The Bases for SR 3.5.3.5 states in part that "this test also ensures the RCIC System will automatically restart on an RPV low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) trip." The requirement in ITS SR 3.5.3.5 is adequate to ensure the RCIC automatic actuation capability is verified to ensure Operability. As such, these details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications. J J

LA2 The details in CTS 4.5.G.3 which describe the method to be employed to assure that the RCIC discharge piping is full of water (shall be vented from the high point of the system and water flow observed) are proposed to be relocated to the Bases. These details are not necessary to ensure the Operability of the RCIC System. The requirements of LCO 3.5.3 (RCIC System) that the RCIC System must be Operable and the associated Surveillances are adequate to ensure the RCIC System remains Operable. Therefore, the relocated details are not required to be in the ITS to

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA2 (continued)

provide adequate protection of the 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 Technical Specifications.

LA3 Not used.

LA4 CTS 4.5.E.1.e requires testable check valve testing for the RCIC System any time the reactor is in the cold shutdown condition exceeding 48 hours if operability tests have not been performed during the preceding 92 days. CTS 4.5.E.1.c requires the RCIC motor operated valves to be tested for Operability every 92 days. These requirements are proposed to be relocated to the TRM. These details are not necessary to ensure Operability of the RCIC System. The requirements of LCO 3.5.3 that the RCIC System must be OPERABLE and SRs 3.5.3.3, 3.5.3.4, and 3.5.3.5 are adequate to ensure the RCIC System remains Operable. Therefore, the relocated requirements are not required to be in the ITS to provide adequate protection of the public health and safety. The TRM will be incorporated by reference in to the JAFNPP UFSAR at ITS Implementation. Changes to the relocated requirements in the TRM will be controlled by the provisions of 10 CFR 50.59. J

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 3.5.E.1 allows continued operation for a maximum of 7 days after RCIC is determined to be inoperable. ITS 3.5.3 Required Action A.2 allows continued operation for a maximum of 14 days under the same conditions. As in the existing Specification, the 14 day Completion Time for restoring RCIC is contingent upon the Operability of HPCI. The 14 day completion time is based on a reliability study that evaluated the impact on ECCS availability (Memorandum from R.L. Baer (NRC) to V. Stello, JR. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975). The main factor contributing to the acceptability of allowing continued operation for 14 days with RCIC inoperable is the similar functions of HPCI and RCIC, and that the HPCI is capable of performing the RCIC function, at a substantially higher capacity.

L2 CTS 3.5.E.2 requires the reactor be in the cold condition and reactor pressure be reduced to less than 150 psig within 24 hours when CTS 3.5.E cannot be met. ITS 3.5.3 Required Actions B.1 and B.2 requires the plant to be in MODE 3 within 12 hours (M1) and to reduce reactor steam dome pressure to ≤ 150 psig within 36 hours under the same conditions. This change is less restrictive since the time to reduce pressure has

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 (continued)

been extend from 24 hours to 36 hours. This change is acceptable since the compensatory action added in accordance with M1 and this extended time to be ≤ 150 psig will ensure a more continuous reduction in power and reactor coolant pressure within the specified maximum cooldown rate and within the capabilities of the plant. The additional time to complete these ACTIONS reduces the potential for a plant event that could challenge plant safety systems.

L3 CTS 4.5.E.1.a stipulates a simulated automatic actuation test shall be performed. The phrase "actual or," in reference to the automatic initiation signal, has been added to CTS 4.5.E.1.a (ITS SR 3.5.3.5) for verifying that each RCIC subsystem actuates on an automatic initiation signal. This allows satisfactory automatic system initiations to be used to fulfill the Surveillance Requirements. Operability is adequately demonstrated in either case since the RCIC System itself can not discriminate between "actual" or "simulated" signals. (J

L4 CTS 4.5.E.2 requires the verification that the HPCI System is Operable immediately and daily thereafter when RCIC is determined to be inoperable. ITS 3.5.3 Required Action A.1 requires immediate verification by administrative means that the HPIC System is Operable, but the explicit requirement for periodic continuing verification has been deleted. These verifications are an implicit part of using Technical Specifications and determining the appropriate Conditions to enter and Actions to take in the event of inoperability of Technical Specification equipment. In addition, plant and equipment status is continuously monitored by control room personnel. The results of this monitoring process are documented in records/logs maintained by control room personnel. The continuous monitoring process includes re-evaluating the status of compliance with Technical Specification requirements when Technical Specification equipment becomes inoperable using the control room records/logs as aids. Therefore, the explicit requirement to periodically verify the Operability of HPCI when RCIC is inoperable is considered to be unnecessary for ensuring compliance with the applicable Technical Specification actions.

L5 CTS 3.5.E.2 requires reactor pressure to be reduced to less than 150 psig. ITS 3.5.3 Required Action B.2 will require reactor pressure be reduced to ≤ 150 psig. This change is slightly less restrictive since a reduction in reactor steam dome pressure to only 150 psig will be considered as satisfying the requirement, whereas in the CTS reactor steam dome pressure must be reduced to < 150 psig. This change is

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L5 (continued)

acceptable since it places the plant outside of the current and proposed Applicability of the RCIC System in CTS 3.5.E.1 (ITS 3.5.3 Applicability). This change is consistent with NUREG-1433, Revision 1.

L6 The CTS 4.5.E.1.d specification that required RCIC flow be demonstrated "against a system head corresponding to a reactor vessel pressure of 1195 to 150 psig" is changed to a demonstration of required RCIC flow "against a system head corresponding to reactor pressure", consistent with NUREG-1433, Revision 1 requirements. The CTS 4.5.E.1.d specification is represented in ITS as two surveillances (see DOC M3), ITS SR 3.5.3.4 performed at a reactor pressure of ≤ 165 psig, and ITS SR 3.5.3.3 performed with reactor pressure ≥ 970 and ≤ 1040 psig. Adopting NUREG wording for ITS SR 3.5.3.4 results in testing requirements analogous to the CTS specification and current testing practice at the low pressure end of the HPCI operability band. Adopting NUREG wording for ITS SR 3.5.3.3 constitutes a less restrictive change.

The RCIC system is designed to provide its rated flow over a reactor pressure range of 150 psig to a maximum pressure based on the lowest SRV safety setpoint. The CTS range of 1195 to 150 psig corresponds to the entire range of operability for RCIC and is intended to demonstrate RCIC operability throughout this range. As noted in DOC M3, however, the CTS does not specify a reactor pressure range for test performance.

In practice, the test is performed at the low end of the range (i.e., ~150 psig) after start-up, and within the normal reactor operating pressure range (970 to 1040 psig) on a periodic basis. CTS testing at the low end of the range demonstrates flow against a discharge head based upon a differential above reactor pressure, consistent with the proposed ITS SR 3.5.3.4. CTS testing in the normal reactor operating pressure range, however, demonstrates flow against a system head derived from the "reactor vessel pressure of 1195" CTS value, not "against a system head corresponding to reactor pressure" as proposed by ITS SR 3.5.3.3.

In actual operation, RCIC system inlet steam pressure and RCIC pump discharge pressure correspond to reactor pressure with allowance for line losses. Requiring that RCIC demonstrate minimum system design flow "against a system head corresponding to a reactor vessel pressure of 1195" with actual reactor steam dome pressure in the normal operating range is overly conservative, since the condition represents less driving steam pressure for the RCIC turbine than would be available if a discharge pressure corresponding to 1195 psig reactor pressure were actually required. RCIC is required to exceed its design operating

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L6 (continued)

requirements to satisfy such test conditions. The NUREG-1433, Revision 1 requirement specifying a reactor pressure range for performing the test and requiring demonstration of flow rate "against a system head corresponding to reactor pressure" constitutes a more accurate and appropriate demonstration of RCIC operability than the CTS in that the NUREG requirements more accurately reflect actual RCIC operating conditions. Since adoption of the NUREG requirements for ITS SR 3.5.3.3 removes a degree of overly restrictive conservatism, the change is considered less restrictive.

TECHNICAL CHANGES - RELOCATIONS

None

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1 [3.5.6] [4.5.6] [4.5.6.3]	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	31 days
[4.5.E.1.b] SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.3.3 [4.5.E.1][M2] [4.5.E.1.d]	------------------NOTE------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with [reactor pressure] ≤ ~~[1020]~~ 1040 psig and ≥ ~~[920]~~ 970 psig, the RCIC pump can develop a flow rate ≥ [400] gpm [against a system head corresponding to reactor pressure]. DB3 DB3 DB2	92 days
SR 3.5.3.4 [4.5.E.1.d] [M3][M2]	------------------NOTE------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with [reactor pressure] ≤ [165] psig, the RCIC pump can develop a flow rate ≥ [400] gpm [against a system head corresponding to reactor pressure]. DB2 DB2	24 X1 ~~[18]~~ months DB3

(continued)

REVISION ~~D~~ J

1 ⑤

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.3.5 [4.5.E 1.a]	----------NOTE ⑤ CLB4---------- 2. Vessel injection may be excluded. -------------------------------- Verify the RCIC System actuates on an actual or simulated automatic initiation signal.	24 CLB3 ~~[18]~~ months

1. Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. CLB4

REVISION ~~0~~ 5

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.5.3 - RCIC SYSTEM

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 Not Used.

CLB2 Not Used.

CLB3 The brackets have been removed and the proper plant specific Frequency has been provided. The Frequency specified in SR 3.5.3.5 of 24 months is consistent with the current requirements in CTS 4.5.E.1.a.

CLB4 A Note has been added to the actual or simulated automatic initiation test in ITS SR 3.5.3.5 (ISTS SR 3.5.3.4) to allow RCIC testing to be delayed until 12 hours after reactor steam dome pressure and flow are adequate. This Note is consistent with the allowances specified in CTS 4.5.E and modified by M3. This modification is necessary to properly test the RCIC pump. The subsequent Note of SR 3.5.3.5 has been renumbered.

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

None

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the proper plant specific value has been provided. The pressure of 150 psig is consistent with the existing requirements in CTS 3.5.E.1 and 3.5.E.2.

DB2 The brackets have been removed and the proper plant specific value/ information has been provided. The 400 gpm flow rate and test pressures specified in ITS SR 3.5.3.3 and 3.5.3.4 are consistent with the current requirements in CTS 4.5.E.1.d.

DB3 The brackets have been removed and the proper plant specific values have been provided. The range of pressures specified in SR 3.5.3.3 (between 970 psig to 1040 psig) are nominal values at rated conditions. The selected pressure condition of ≤ 165 psig in SR 3.5.3.4 is very close to the lower range where RCIC is required to be Operable, however, at the same time allows some flexibility to establish the condition.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.3 RCIC System

BASES

BACKGROUND

The RCIC System is not part of the ECCS; however, the RCIC System is included with the ECCS section because of their similar functions.

The RCIC System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of the RPV water level. Under these conditions, the High Pressure Coolant Injection (HPCI) and RCIC systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 1 are satisfied.

PA1

The RCIC System (Ref. 2) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping is provided from the condensate storage tanks (CSTs) and the suppression pool. Pump suction is normally aligned to the CSTs to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, ~~or the suppression pool level is high,~~ an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from a main steam line upstream of the associated inboard main steam line isolation valve.

J DB1

the "B"

The RCIC System is designed to provide core cooling for a wide range of reactor pressures ~~165~~ 150 psig to ~~1155~~ 1195 psig. Upon receipt of an initiation signal, the RCIC turbine accelerates to a specified speed. As the RCIC flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water ~~from and~~ to the CST to allow testing of the RCIC System during normal operation without injecting water into the RPV.

DB2 DB1 PA2

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.2 (continued)

31 days is further justified because the valves are operated under procedural control and because improper valve position would affect only the RCIC System. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.3.3 and SR 3.5.3.4

DB1 against a system head corresponding to reactor pressure

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow is tested both at the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Reactor steam pressure must be ≥ ~~920~~ 970 psig to perform SR 3.5.3.3 and ≥ ~~150~~ psig to perform SR 3.5.3.4. Adequate steam flow is represented by at least ~~1.25~~ one turbine bypass valves open, or ~~total steam flow ≥ 10^6 lb/hr~~. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these SRs. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure Surveillance has been satisfactorily completed and there is no indication or reason to believe that RCIC is inoperable. ~~Therefore~~, these SRs are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test.

DB6 · PA2 Insert SR 3-B · PA2 Adequate · DB6 main turbine generator load is greater than 100 MWe · PA2 · A · PA2 · J

Insert SR 3-C PA2

A 92 day Frequency for SR 3.5.3.3 is consistent with the Inservice Testing Program requirements. The ~~18~~ 24 month Frequency for SR 3.5.3.4 is based on the need to perform the Surveillance under conditions that apply ~~just prior to or~~ during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the ~~18~~ 24 month Frequency, which is based on the

X2 · J · PA2 · 24 X2

(continued)

1 (J)

PA 2

INSERT SR3-B

The required system head should overcome the RPV pressure and associated discharge line losses. Adequate reactor steam pressure must be available to perform these tests.

PA 2

INSERT SR3-C

The 12 hours allowed for performing the flow test after the required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides reasonable time to complete the SR.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.3 and SR 3.5.3.4 (continued)

refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.3.5

The RCIC System is required to actuate automatically in order to verify its design function satisfactorily. This Surveillance verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of the RCIC System will cause the system to operate as designed, including actuation of the system throughout its emergency operating sequence; that is, automatic pump startup and actuation of all automatic valves to their required positions. This test also ensures the RCIC System will automatically restart on an RPV low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) ~~trip~~ and that the suction is automatically transferred from the CST to the suppression pool. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.2 overlaps this Surveillance to provide complete testing of the assumed ~~safety~~ design function. PA2

DB1

signal (Level 8 signal closes RCIC steam inlet valve, and subsequent Level 2 signal will reopen valve)

The ~~18~~ 24 month Frequency is based on the need to perform the 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~~ startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the ~~18~~ 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. CLB3

This SR is modified by ~~a~~ 2 Note that excludes vessel injection during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

Insert SR 3.5.3.5

(continued)

REVISION ~~0~~ J

INSERT SR 3.5.3.5 CLB3

1

This SR is modified by Note 1 that says the Surveillance is not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The time allowed for this test after required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. Adequate reactor pressure must be available to perform this test. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Thus, sufficient time is allowed after adequate pressure and flow are achieved to perform this test. Adequate reactor steam pressure is > 150 psig. Adequate steam flow is represented by at least one turbine bypass valve open. Reactor startup is allowed prior to performing this test because the reactor pressure is low and the time allowed to satisfactorily perform the test is short.

BASES (continued)

DB4

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 33~~ UFSAR, Section 16.6. 1J

PA3

2. UFSAR, Section ~~[5.5.6]~~ 4.7 DB5 1J

~~3~~ 4 Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

X1 PA4

3 10 CFR 50.36(c)(2)(ii). X1 1J

REVISION ~~0~~ J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.5.3 - RCIC SYSTEM

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 Not Used.

CLB2 Not Used.

CLB3 The 18 month Frequency has been changed to 24 months consistent with the current fuel cycle. This Frequency specified in SR 3.5.3.5 of 24 months is consistent with the current requirements in CTS 4.5.E.1.a. The Bases have been revised to reflect the plant specific design and justification.

CLB4 A Note has been added to the actual or simulated automatic initiation test in ITS SR 3.5.3.5 to allow HPCI testing to be delayed until 12 hours after reactor steam dome pressure and flow are adequate. This Note is consistent with the allowances specified in CTS 4.5.E and modified by M2. This modification is necessary to properly test the RCIC pump. The subsequent Note of SR 3.5.3.5 has been renumbered. The Bases has been modified as required to reflect this modification.

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

PA1 Editorial changes have been made to correct typographical error.

PA2 Editorial changes have been made for enhanced clarity with no change in intent.

PA3 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.

PA4 The quotations used in the Bases References have been removed. The Writer's Guide does not require the use of quotations.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.5.3 - RCIC SYSTEM

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 brackets have been removed and the proper plant specific design values/information have been provided.

DB3 The brackets have been removed and the proper plant specific value has been provided. The pressure of 150 psig is consistent with the existing requirements in CTS 3.5.E.1 and 3.5.E.2.

DB4 JAFNPP was designed and under construction prior to the promulgation of Appendix A to 10 CFR 50 - General Design Criteria for Nuclear Power Plant. The JAFNPP Construction Permit was issued on May 20, 1970. The proposed General Design Criteria (GDC) were published in the Federal Register on July 11, 1967 (32 FR 10213) and became effective on February 20, 1971 (32 DR 3256). UFSAR Section 16.6 - Conformance to AEC Design Criteria, describes the JAFNPP current licensing basis with regard to the GDC. ISTS statements concerning the GDC are modified in the ITS to reference UFSAR Section 16.6.

DB5 The brackets have been removed and the proper plant specific Reference has been provided.

DB6 The brackets have been removed and the proper plant specific values/ information have been provided. The range of pressures specified in SR 3.5.3.3 (between 970 psig to 1040 psig) are nominal values at rated conditions and therefore are appropriate for this test. The selected pressure condition of $\leq$ 165 psig in SR 3.5.3.4 is very close to the lower range where RCIC is required to be Operable, however, at the same time allows some flexibility to establish the condition. The Bases has been modified as required to reflect these changes 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 301, Revision 0, have been incorporated into the revised Improved Technical Specifications.

TA2 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 367, Revision 0, have been incorporated into the revised Improved Technical Specifications.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.3.3	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 1040 psig and ≥ 970 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.3.4	---------------NOTE--------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 165 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.3.5	-----NOTES----- 1. Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. 2. Vessel injection may be excluded. ----- Verify the RCIC System actuates on an actual or simulated automatic initiation signal.	24 months

BASES (continued)

ACTIONS

A.1 and A.2

If the RCIC System is inoperable during MODE 1, or MODE 2 or 3 with reactor steam dome pressure > 150 psig, and the HPCI System is verified to be OPERABLE, the RCIC System must be restored to OPERABLE status within 14 days. In this Condition, loss of the RCIC System will not affect the overall plant capability to provide makeup inventory at high reactor pressure since the HPCI System is the only high pressure system assumed to function during a loss of coolant accident (LOCA). OPERABILITY of HPCI is therefore verified immediately when the RCIC System is inoperable. This may be performed as an administrative check, by examining logs or other information, to determine if HPCI is out of service for maintenance or other reasons. It does not mean it is necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the HPCI System. If the OPERABILITY of the HPCI System cannot be verified, however, Condition B must be immediately entered. For transients and certain abnormal events with no LOCA, RCIC (as opposed to HPCI) is the preferred source of makeup coolant because of its relatively small capacity, which allows easier control of the RPV water level. Therefore, a limited time is allowed to restore the inoperable RCIC to OPERABLE status.

The 14 day Completion Time is consistent with the recommendations in a reliability study (Ref. 4) that evaluated the impact on ECCS availability, assuming various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (AOTs). Because of similar functions of HPCI and RCIC, the AOTs (i.e., Completion Times) determined for HPCI are also applied to RCIC.

B.1 and B.2

If the RCIC System cannot be restored to OPERABLE status within the associated Completion Time, or if the HPCI System is simultaneously inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to ≤ 150 psig within 36 hours. The allowed Completion Times

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.2 (continued)

31 days is further justified because the valves are operated under procedural control and because improper valve position would affect only the RCIC System. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.3.3 and SR 3.5.3.4

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow against a system head corresponding to reactor pressure is tested both at the higher and lower operating ranges of the system. The required system head should overcome the RPV pressure and associated discharge line losses. Adequate reactor steam pressure must be available to perform these tests. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these SRs. Adequate reactor steam pressure must be ≥ 970 psig to perform SR 3.5.3.3 and > 150 psig to perform SR 3.5.3.4. Adequate steam flow is represented by at least one turbine bypass valve open, or main turbine generator load is greater than 100 MWe. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure Surveillance has been satisfactorily completed and there is no indication or reason to believe that RCIC is inoperable.

These SRs are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for performing the flow test after the required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides reasonable time to complete the SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.3 and SR 3.5.3.4 (continued)

A 92 day Frequency for SR 3.5.3.3 is consistent with the Inservice Testing Program requirements. The 24 month Frequency for SR 3.5.3.4 is based on the need to perform the Surveillance under conditions that apply during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.3.5

The RCIC System is required to actuate automatically in order to verify its design function satisfactorily. This Surveillance verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of the RCIC System will cause the system to operate as designed, including actuation of the system throughout its emergency operating sequence; that is, automatic pump startup and actuation of all automatic valves to their required positions. This test also ensures the RCIC System will automatically restart on an RPV low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) signal (Level 8 signal closes RCIC steam inlet valve, and subsequent Level 2 signal will re-open valve) and that the suction is automatically transferred from the CST to the suppression pool. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.2 overlaps this Surveillance to provide complete testing of the assumed design function.

The 24 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by Note 1 that says the Surveillance is not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The time allowed for this test after required

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.5 (continued)

pressure and flow are reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. Adequate reactor pressure must be available to perform this test. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Thus, sufficient time is allowed after adequate pressure and flow are achieved to perform this test. Adequate reactor steam pressure is > 150 psig. Adequate steam flow is represented by at least one turbine bypass valve open. Reactor startup is allowed prior to performing this test because the reactor pressure is low and the time allowed to satisfactorily perform the test is short.

This SR is modified by Note 2 that excludes vessel injection during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

REFERENCES

1. UFSAR, Section 16.6.
2. UFSAR, Section 4.7.
3. 10 CFR 50.36(c)(2)(ii).
4. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), Recommended Interim Revisions to LCOs for ECCS Components, December 1, 1975.