# ATTACHMENT 1

| Current Number | Title | STS Rev. 4 Number | New TS Number | Retained/ Criterion for Inclusion | Bases for Inclusion/Exclusion[a] |
|---|---|---|---|---|---|
| APPENDIX B | RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS | | | | |
| 1.0 | Definitions | None | 1.1 | Yes | See Note 1, Page 14. |
| 2.0 | Liquid Effluents | | | | |
| 2.1 | Liquid Effluent Monitors (Table 2.1-1 and 3.10-2) | None | Relocated | No | See Appendix A, Page 10. |
| 2.2 | Concentration of Liquid Effluents (Table 2.1-1) | None | Relocated | No | See Appendix A, Page 11. |
| 2.3 | Dose from Liquid Effluents | None | Relocated | No | See Appendix A, Page 12. |
| 2.4 | Liquid Radioactive Waste Treatment System | None | Relocated | No | See Appendix A, Page 13. |
| 2.5 | Maximum Activity in Outside Tanks | None | 5.5.8 | Yes | Although this Specification does not meet any criteria of the NRC Final Policy Statement, it has been retained in accordance with the NRC letter from W.T. Russell to the industry ITS Chairpersons dated October 25, 1993. |
| 3.0 | Gaseous Effluents | | | | |
| 3.1 | Gaseous Effluent Monitors (Table 3.10-1 and 3.10-2) | None | Relocated | No | See Appendix A, Page 14. |
| 3.2 | Gaseous Dose Rate (Table 3.2-1) | None | Relocated | No | See Appendix A, Page 15. |
| 3.3 | Air Dose, Noble Gases | None | Relocated | No | See Appendix A, Page 16. |
| 3.4 | Dose due to Iodine-131, Tritium, and Radionuclides in Particulate Form | None | Relocated | No | See Appendix A, Page 17. |
| 3.5.a | Main Condenser Steam Jet Air Ejectors (SJAE) (Table 3.10-1 and 3.10-2) | None | 3.7.5 | Yes-2 | Main condenser offgas activity is an initial condition in the offgas system failure event. (J) |
| 3.5.b | SJAE Radiation Monitors (Tables 3.10-1 and 3.10-2) | None | Relocated | No | See Appendix A, Page 23a. (J) |
| 3.6 | Offgas Treatment System | None | Relocated | No | See Appendix A, Page 18. |

(a) The applicable safety analyses are discussed in the Bases for the individual Technical Specification.

3.5.b SJAE RADIATION MONITORS

LCO Statement:

Except as specified in 1. and 2. below, both SJAE system radiation monitors shall be operable during reactor power operation. The trip time delay setting for closure of the SJAE isolation valve shall not exceed 15 min.

Discussion:

The SJAE radiation monitors are neither a safety system nor are they connected to the reactor coolant system. The primary function of this instrumentation is to show conformance to the discharge limits of 10 CFR Part 20. This instrumentation is not installed to detect excessive reactor coolant leakage. The SJAE radiation monitors are used routinely to provide a continuous check on the releases of radioactive gaseous effluents from the Main Condenser steam jet air ejector. These Technical Specifications require the Licensee to maintain operability of the SJAE radiation monitors and establish a setpoint. The alarm/trip setpoint are established to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. Plant DBA analyses do not assume any action, either automatic or manual, resulting from these monitors.

Comparison to Screening Criteria

1. The SJAE radiation monitors are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The SJAE radiation monitors are not used to monitor a process variable that is an initial condition of a DBA or transient. J

3. The SJAE radiation monitors are not part of a primary success path in the mitigation of a DBA or transient. Excessive discharge is not considered to initiate a primary success path in mitigating a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (item 189) of NEDO-31466, the loss of the SJAE radiation monitors was found to be a non-significant risk contributor to core damage frequency and offsite releases. ENO has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisified, the SJAE Radiation Monitors LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

SUMMARY OF CHANGES TO ITS CHAPTER 1.0 - 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 ITS markup. (The word "met" has been changed to "performed" and the word "three" has been changed to "there".) | **Specification 1.4**<br><br>Retyped ITS p 1.4-7 |
| NUREG ITS markup errors | Minor NUREG markup errors have been corrected to be consistent with the retyped ITS. (A comma has been deleted and a period has been added to the CHANNEL FUNCTIONAL TEST definition; and a comma has been deleted, a comma has been added, and a period has been added to the DOSE EQUIVALENT I-131 definition.) | **Specification 1.1**<br><br>NUREG ITS markup p 1.1-2, 1.1-3, and Insert page 1.1-3 |
| Typographical errors | Minor typographical errors have been corrected. (The word "satified" has been changed to "satisfied" and the word "require" has been changed to "required".) | **Specification 1.4**<br><br>NUREG ITS markup p Insert page 1.4-2<br><br>Retyped ITS p 1.4-2 |
| Consistency issue | The word "assumed" has been added to the Turbine Bypass System Response Time definition, since the turbine bypass capacity referenced in the definition is from only three of the valves, not all four. See the Consistency change issue in the Summary for ITS Section 3.7 for further details. | **Specification 1.1**<br><br>NUREG markup p 1.1-7<br><br>JFD X3 (JFDs p 5 of 5)<br><br>Retyped ITS p 1.1-6 |

CTS

## 1.1 Definitions

| | | |
|---|---|---|
| | CHANNEL CHECK (continued) | status derived from independent instrument channels measuring the same parameter. |
| [1.0.F.5] | CHANNEL FUNCTIONAL TEST | A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested. |
| [1.0.B] | CORE ALTERATION | CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:<br><br>a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and<br><br>b. Control rod movement, provided there are no fuel assemblies in the associated core cell.<br><br>Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position. |
| [1.0.AD] | CORE OPERATING LIMITS REPORT (COLR) | The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications. |
| [RETS 1.0.A] | DOSE EQUIVALENT I-131 | DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose |

TA2: of all devices in the channel required for channel OPERABILTY

PA1

J

(continued)

CTS

## 1.1 Definitions

RETS 1.0.A

**DOSE EQUIVALENT I-131 (continued)**

INSERT 1.1-1

CLB7

conversion factors used for this calculation shall be those listed in ~~[Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites" or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity"].~~

CLB7

J

**EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME**

The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

CLB1

**END OF CYCLE RECIRCULATION PUMP TRIP (EOC RPT) SYSTEM RESPONSE TIME**

The EOC RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by [the associated turbine stop valve limit switch or from when the turbine control valve hydraulic oil control oil pressure drops below the pressure switch setpoint] to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured, [except for the breaker arc suppression time, which is not measured but is validated to conform to the manufacturer's design value].

DB1

CLB2

INSTRUMENTATION

[1.0.F.6]

**ISOLATION SYSTEM RESPONSE TIME**

The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel ~~to their required positions. Times shall include diesel generator starting and sequence loading delays, where applicable.~~ The response time may be measured by means of any series of sequential,

CLB2

receives the isolation signal (e.g., de-energization of the main steam isolation valve solenoids).

(continued)



REVISION ~~0~~ J

INSERT 1.1-1

... International Commission on Radiological Protection Publication 30 (ICRP-30), "Limits for Intake by Workers," or in ... 1(J)

## 1.1 Definitions

| | |
|---|---|
| SHUTDOWN MARGIN (SDM) (continued) | With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM. |
| [1.0.X] STAGGERED TEST BASIS | A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function. |
| [A15] THERMAL POWER | THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant. |
| [A15] TURBINE BYPASS SYSTEM RESPONSE TIME | The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components:<br>a. The time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established; and<br>b. The time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve.<br>The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. |

assumed

X3

J

INSERT 1.4-1

Some Surveillances contain notes that modify the Frequency of performance of the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; or

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The definition of EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME does not exist in the CTS and is not adopted in the ITS. Since no requirements for ECCS response time testing exist in the CTS, no definition is necessary. The mechanical (valve) portion of the response time (i.e., valve stroke times) does not need to be included since it is redundant to the valve stroke time requirements specified in ASME Section XI, which is required by proposed Specification 5.5.7, the IST Program. In addition, generic studies have shown that instrumentation response time changes (increasing times), that could impact safety, do not normally vary such that they would not be detected during other required surveillances (e.g., Channel Calibrations). Since the addition of these tests would be a major burden, with little gain in safety, the SR's associated with these tests have not been added for any ECCS instrumentation.

CLB2 The ISOLATION SYSTEM RESPONSE TIME definition has been modified to only include the instrumentation portion of the response time. The isolation valve portion of the response time (i.e., valve stroke times) does not need to be included since it is redundant to the valve stroke time requirements specified in ASME Section XI, which is required by proposed Specification 5.5.7, the IST Program. In addition, specific Surveillance Requirements in LCO 3.6.1.3, Primary Containment Isolation Valves, also require the valve stroke times to be verified. The requirement to include diesel generator starting and loading times has been deleted since they are redundant to the diesel generator Surveillance Requirements in LCO 3.8.1, AC Sources - Operating. This deletion was recommended in both NUREG-1366 and Generic Letter 93-05. Due to these changes, the definition has been renamed to be ISOLATION INSTRUMENTATION RESPONSE TIME. The definition of ISOLATION SYSTEM RESPONSE TIME is modified to reflect that response time testing is not required for any of the isolation systems and associated isolation actuation instrumentation, with the exception of the main steam isolation valves (MSIVs); and that the response time is the interval from the time when the monitored parameter exceeds its setpoint until the MSIV solenoids are de-energized, consistent with current licensing basis. These changes are consistent with the CTS 1.0.F.6 definition for isolation instrumentation response time and CTS 4.2.A footnote * as previously approved. The footnote in CTS 4.2.A which provides an allowance to exclude testing for sensors is based on the requirements of NEDO-32291-A and Supplement 1 as documented in Amendment 235.

CLB3 Not Used

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB4 The ISTS definition for MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) has been retained in the JAFNPP consistent with the current licensing basis described in CTS 1.0.U.2 and 1.0.U.3.

CLB5 The ITS 1.1 definition for RATED THERMAL POWER (RTP) has been revised to reflect the JAFNPP current licensing basis value of 2536 MWt, as indicated in CTS 1.0.N.

CLB6 The ITS 1.1 Table 1.1-1, MODES, has been revised to reflect the JAFNPP current licensing basis values, of > 212°F for Hot Shutdown and ≤ 212°F for Cold Shutdown, as indicated in CTS 1.0.I.3.a and 1.0.I.3.b.

CLB7 The brackets have been removed and the thyroid dose conversion factors used will be those listed in NRC Regulatory Guide 1.109 or ICRP-30 consistent with the current definition.

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

PA1 Editorial changes have been made for enhanced clarity or to correct a grammatical/typographical error.

PA2 The definition of MINIMUM CRITICAL POWER RATIO (MCPR) is modified to refer to "type" of fuel, rather than "class" of fuel, consistent with plant specific terminology.

PA3 NUREG-1433, Revision 1, ISTS 1.3, Example 1.3-3 and Example 1.3-6 (ITS Example 1.3-3 and Example 1.3-6) are revised to more adequately reflect JAFNPP specific Technical Specifications ACTIONS rather than PWR specific Technical Specifications ACTIONS.

In Example 1.3-3, the Completion Times for Condition C are revised from "72 hours" to "12 hours." The JAFNPP ITS does not contain any Conditions similar to Example 1.3-3 Condition C; where the Completion Time for restoring multiple LCO requirements in a separate Condition is the same as the Completion Time for restoring a single LCO requirement. The discussion for Example 1.3-3 explains how multiple Condition entry works in the example, how separate Completion Times are tracked for each Condition in the example, and the proper application of the maximum Completion Time in Conditions A and B of the example. However, no Completion Times are ever made, with the exception of the maximum Completion Time, in the discussion of Example 1.3-3. The maximum Completion Time is not modified by this change.

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

PA3 (continued)

In Example 1.3-6, Required Action A.2 is revised from "Reduce THERMAL POWER to ≤ 50% RTP" to "Place channel in trip." The JAFNPP ITS does not contain any Conditions similar to Example 1.3-6 Condition A; where optional Required Actions exist for an instrument channel inoperable and one of the Required Actions is to reduce power. The discussion for Example 1.3-6 explains how multiple Condition entry works in the example, how the logical connector works in the example, and the proper application of the Completion Time for Required Actions A.1 and A.2 of the example. However, no reference to the specific details of the Required Actions is ever made in the discussion of Example 1.3-6. Specific references to the Completion Times are made in the discussion of Example 1.3-6. The Completion Times are not modified by this change.

The examples in Specification 1.3 are provided to help ensure the Completion Time convention in the JAFNPP ITS is understood and properly applied. No changes are required to the existing discussions of the examples as a result of these changes to the examples. Therefore, these changes do not impact the discussions of the associated examples. The changes are only to make the examples JAFNPP specific. As a result, the changes do not impact the examples' use in helping to ensure the ITS Completion Time convention is understood and properly applied.

PA4 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.

PA5 Changes have been made to be consistent with other places in the ITS.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The definition of END OF CYCLE RECIRCULATION PUMP TRIP (EOC RPT) SYSTEM RESPONSE TIME is not adopted in the ITS. The EOC RPT feature is not part of the JAFNPP design.

DB2 The brackets have been removed and the ISTS 1.1 definition for LINEAR HEAT GENERATION RATE (LHGR) has been included at JAFNPP since ISTS 3.2.3, "LINEAR HEAT GENERATION RATE", is retained and the definition is used in other places in the Bases.

DB3 The brackets have been removed from the APLHGR definition and the definition retained as described in the NUREG. Reference to LHGR has been deleted since it is defined in a different definition. The definition is consistent with current interpretation of the use of the

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB3 (continued)

term in the CTS.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 52, Revision 3, have been incorporated into the revised Improved Technical Specifications.

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

TA3 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 284, Revision 3, have been incorporated into the revised Improved Technical Specifications.

TA4 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 332, Revision 1, have been incorporated into the revised Improved Technical Specifications.

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

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

X1 The ISTS 1.1 definition for PHYSICS TESTS is not retained at JAFNPP since it is not used. NUREG-1433, Revision 1, ISTS 3.10.9, "Recirculation Loops-Testing," referring to the PHYSICS TESTS definition, is not included in the JAFNPP ITS. The justification for differences from NUREG-1433, Revision 1, for ITS 3.10, Special Operations, addresses not including ISTS 3.10.9 in the JAFNPP ITS.

X2 The ISTS 1.1 definition for RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) is not retained in the JAFNPP ITS since NRC approved methodology for the development of RCS pressure and temperature limits does not exist at JAFNPP.

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

X3 The ISTS 1.1 definition for TURBINE BYPASS SYSTEM RESPONSE TIME has been included, consistent with NUREG-1433, Revision 1, with the exception of the word "assumed". The word has been added since only three of the four main turbine bypass valves are assumed to open. This is described in the Bases for ITS 3.7.6 Main Turbine Bypass System.

5

## 1.1 Definitions (continued)

| | |
|---|---|
| THERMAL POWER | THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant. |
| TURBINE BYPASS SYSTEM RESPONSE TIME | The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components:<br><br>a. The time from initial movement of the main turbine stop valve or control valve until 80% of the assumed turbine bypass capacity is established; and<br><br>b. The time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve.<br><br>The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. |

1 J

## 1.4 Frequency

DESCRIPTION (continued)

criteria. Some Surveillances contain notes that modify the Frequency of performance of the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; or

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

(continued)

## 1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| ----------NOTE---------- Only required to be performed in MODE 1. ---------- | |
| Perform complete cycle of the valve. | 7 days |

The interval continues, whether or not the plant operation is in MODE 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the plant reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance were not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

(continued)

SUMMARY OF CHANGES TO ITS CHAPTER 2.0 - REVISION J

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Typographical error | A minor typographical error has been corrected. (The term "Loss of Coolant" has been changed to "Loss-of-Coolant".) | Specification 2.1.1<br>NUREG Bases markup p B 2.0-7<br>Retyped ITS Bases p B 2.0-5 |
| Consistency issue | A minor consistency issue correction has been made. (The revision and date of Reference 2, which is also identified in the COLR, has been deleted and a statement that the Revision is specified in the COLR has been added. This is consistent with Specification 5.6.5.) | Specification 2.1.1<br>NUREG Bases markup p B 2.0-7<br>Retyped ITS Bases p B 2.0-5 |

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.3

If any SL is violated, the [senior management of the nuclear plant and the utility Vice President—Nuclear Operations] shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the appropriate utility management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 6). A copy of the report shall also be provided to the [senior management of the nuclear plant and the utility Vice President—Nuclear Operations].

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 10.~~ VFSAR, Section 16.6 DB4
2. NEDE-24011-P-A ~~(latest approved revision)~~. DB1
3. ~~XN-NF524(A), Revision 1, November 1983.~~ DB3 General Electric Standard Application for Reactor Fuel, (Revision specified in the COLR)
4. ~~10 CFR 50.72.~~
4 ~~5.~~ 10 CFR 100.
~~6. 10 CFR 50.73.~~
3. NEDC-31317P, Revision 2, James A. Fitz Patrick Nuclear Power Plant SAFER/GESTR-LOCA, Loss-of-Coolant Accident Analysis, April 1993. DB2

DB1 numbering

TA1



REVISION D

BASES (continued)

SAFETY LIMIT VIOLATIONS

Exceeding a SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. UFSAR, Section 16.6.

2. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, (Revision specified in the COLR). J

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

4. 10 CFR 100.

SUMMARY OF CHANGES TO ITS SECTION 3.0 - REVISION J

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Consistency issue | The ITS Bases LCO 3.0.3 words "Specification 3.0.3" have been changed to "LCO 3.0.3" to be consistent with its usage throughout this section. | **Specification 3.0**<br><br>NUREG Bases markup p B 3.0-4<br><br>Retyped ITS Bases p B 3.0-4 |
| Consistency issue | A statement in the Bases for ITS 3.0.4 has been deleted since it was in conflict with the philosophy of the ITS. Specifically, the statement in the Bases said that even if an SR was precluded form being performed during a certain Mode, it could be credited if the SR were unplanned. Since the Bases cannot change the Technical Specification requirements (which in this case it does), this statement has been deleted. | **Specification 3.0**<br><br>NUREG Bases markup p Insert Page B 3.0-10<br><br>JFD TA2 (JFDs p 2 of 3)<br><br>Retyped ITS Bases p B 3.0-12 |
| Editorial change | A portion of the A DOC should have been deleted in the Revision D submittal. Amendment 262 approved the change discussed in the second paragraph of the DOC, so it is now in the CTS and is not added as part of the ITS submittal. Therefore, this second paragraph has been deleted. | **Specification 3.0**<br><br>DOC A12 (DOCs p 5 of 9) |
| TSTF-358 | The changes approved in TSTF-358, concerning SR 3.0.3, have been made. | **Specification 3.0**<br><br>DOCs L3 and L4 (DOCs p 7 of 9, 8 of 9, and 9 of 9)<br><br>NSHCs L3 and L4 (NSHCs p 4 of 7, 5 of 7, 6 of 7, and 6 of 7)<br><br>NUREG ITS markup p 3.0-4<br><br>JFD TA4 (JFDs p 1 of 2)<br><br>NUREG Bases markup p B 3.0-13 and Insert Page B 3.0-13<br><br>Bases JFD TA8 (Bases JFDs p 2 of 3)<br><br>Retyped ITS p 3.0-4<br><br>Retyped ITS Bases p B 3.0-15 and B 3.0-16 |

DISCUSSION OF CHANGES
ITS 3.0 - LCO AND SR APPLICABILITY

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 3.0.A states that the LCOs and Actions shall be applicable during the OPERATIONAL CONDITIONS (modes) specified for each specification. ITS LCO 3.0.1 replaces the CTS phrase "Limiting Conditions for Operation...shall be applicable..." with the phrase "LCOs shall be met..." This change is made to be consistent with the format of other LCO 3.0 Specifications and with the concept of an LCO being met. In addition ITS LCO 3.0.1 identifies specific exceptions to other LCO Applicabilities thus eliminating any interpretations that may be required, and avoiding any confusion. These changes constitute editorial rewording, and presentation preferences consistent with the BWR/4 ISTS, NUREG-1433, Revision 1, and are administrative.

A3 CTS 3.0.B states that the LCO is complied with if the Actions are completed (within the specified time interval) or if the LCO is restored prior to the time interval expiring. ITS LCO 3.0.2 rewords the current requirement to be consistent with the format of other LCO 3.0 Specifications. In addition ITS LCO 3.0.2 identifies specific exceptions to other LCO Applicabilities thus eliminating any interpretations that may be required, and avoiding any confusion. These changes constitute editorial rewording, and presentation preferences consistent with the BWR/4 ISTS, NUREG-1433, Revision 1, and are administrative.

A4 A phrase has been added to CTS LCO 3.0.C for clarity. ITS LCO 3.0.3 includes the phrase "LCO 3.0.3 is only applicable in MODES 1, 2, and 3." This phrase has been added since CTS provides no guidance in this area. No further ACTIONS would be required to be performed if the plant were already in MODE 4 or 5 since CTS LCO 3.0.C only requires the plant to be placed in MODE 4. This change constitutes editorial rewording, and presentation preferences consistent with the BWR/4 ISTS, NUREG-1433, Revision 1, and is administrative.

A5 Two CTS Surveillance Requirements, 4.0.A and 4.0.C have been combined to form ITS SR 3.0.1. ITS SR 3.0.1 rewords the current requirements to be consistent with the format of other LCO 3.0 Specifications. ITS SR 3.0.1 also adds clarifying words specifying that "failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO." CTS implies that failure to meet the Surveillance means failure to meet the LCO, however ITS SR 3.0.1 provides this information in a clearer manner. This change constitutes editorial rewording, and presentation preferences consistent with the BWR/4 ISTS, NUREG-1433, Revision 1, and are administrative.

A6 CTS 4.0.B allows the Surveillance Frequency to be extended by 25% each Surveillance interval. ITS SR 3.0.2 rewords the current requirement to be consistent with the format of other LCO 3.0 Specifications. ITS 3.0.2 also adds the sentence "Exceptions to this Specification are stated in the individual Specifications," to acknowledge the explicit use of exceptions in various Surveillances. The basic application of the 25% extension to routine Surveillances is maintained. These changes constitute editorial rewording, and presentation preferences consistent with the BWR/4 ISTS, NUREG-1433, Revision 1, and are administrative.

A7 When it is determined that a Surveillance Requirement has not been performed, CTS 4.0.C provides allowances for delay into the ACTIONS requirements for up to 24 hours for those specifications which include out of service times of less than 24 hours. This allowance has been modified as described in L1. CTS 4.0.C has been revised to explicitly state the required ACTIONS if the Surveillance is not performed within the delay period or if the Surveillance is performed within the delay period but it is not met. The second paragraph of ITS SR 3.0.3 requires the LCO to be immediately declared not met, and the applicable Condition(s) to be entered if the Surveillance is not performed within the delay period. The third paragraph requires these same actions when the Surveillance is performed within the delay period but is not met. Since the actions are implied in CTS 4.0.C, this change is considered administrative. This change is consistent with NUREG-1433, Revision 1.

A8 CTS 3.0.D does not permit entry into a MODE or other specified condition when an LCO is not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Exceptions to these requirements are stated in the individual specifications. ITS LCO 3.0.4 rewords the current requirement to be consistent with the format of other LCO 3.0 Specifications.

In addition, ITS LCO 3.0.4 states that the LCO is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3. A review of the current and proposed Specifications has been performed to determine the affects of this allowance on the current and proposed Specifications. The review has determined that this change does not provide any additional allowances to change MODES beyond those that currently exist, except where justified in individual Specifications (as described in the individual Specification's Discussion of Changes). These changes constitute editorial rewording, and presentation preferences consistent with the BWR/4 ISTS, NUREG-1433, Revision 1, and are administrative.

A9 CTS 3.0.F states that this LCO is an exception to LCO 3.0.B (ITS 3.0.2). ITS LCO 3.0.5 includes these requirements and also adds clarifying words specifying that the exception to LCO 3.0.2 is for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY. This clarification eliminates any interpretations that may be required, and avoids any confusion. These changes constitute editorial rewording, and presentation preferences consistent with the BWR/4 ISTS, NUREG-1433, Revision 1, and are administrative.

A10 ITS LCO 3.0.6 is added to provide guidance regarding the appropriate actions to be taken when a single inoperability (e.g., a support system) also results in the inoperability of one or more related systems (e.g., supported system(s)). The existing Technical Specifications and various NRC guidance documents have not provided a consistent approach to the combined support/supported inoperability.

- Guidance provided in the June 13, 1979, NRC memorandum from Brian K. Grimes (Assistant Director for Engineering and Projects) to Samuel E. Bryan (Assistant Director for Field Coordination)

indicates an intent/interpretation consistent with the proposed LCO 3.0.6 - without the necessity of also requiring the additional actions of a Safety Function Determination Program. That is, only the inoperable support system actions need be taken.

- Guidance provided by the NRC in their April 10, 1980, letter to all Licensees regarding the definition of Operability and the impact of a support system on the remainder of the Technical Specifications, indicates a similar philosophy of not taking actions for the inoperable supported equipment. However, in this case, additional actions similar to the proposed Safety Function Determination Program actions, were addressed and required.

- Generic Letter 91-18 and a literal reading of the existing Standard Technical Specifications provide the interpretation that failure to perform a required function, even as a result of an inoperable Technical Specification support system, requires all associated actions be taken.

- Certain existing specifications contain actions such as: Declare the supported system inoperable and take the Actions of its specification. In many cases the supported system would already be considered inoperable. The implication of this presentation is that the actions of the inoperable supported system would not have been taken without the specific action to do so.

Considering the history of disagreement and misunderstandings in this area, the ISTS were developed with Industry input and approval of the NRC to include ITS LCO 3.0.6. Since its function is to clarify existing ambiguities and maintain actions within the realm of previous interpretations, this new provision, consistent with the BWR/4 ISTS, NUREG-1433, Revision 1, is deemed to be administrative in nature.

A11 A requirement has been added to CTS 3.0.C (ITS LCO 3.0.3) which requires entry into LCO 3.0.3 when directed by the associated ACTIONS. This requirement is not included in the CTS since no specification explicitly directs entry into CTS 3.0.C. Since the ITS also uses this method of entry into LCO 3.0.3 this statement must be included. Changes to Specifications to explicitly require direct entry into LCO 3.0.3 (e.g., ITS 3.5.1) in the ITS if certain conditions are not met, are discussed in the Discussion of Changes for the specific Specification. Therefore,

ADMINISTRATIVE CHANGES

A11 (continued)

this change constitutes a presentation preference consistent to NUREG-1433, Revision 1 and is considered administrative.

A12 CTS 4.0.D does not permit entry into a MODE or other specified condition when an LCO's Surveillances have not been met within the applicable interval or as otherwise stated. ITS SR 3.0.4 rewords the current requirement to be consistent with the format of other LCO 3.0 Specifications.

In addition, ITS SR 3.0.4 states that the SR is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3. This phrase has been added since CTS 4.0.D provides no guidance in this area. The change eliminates the restrictions of the SR when in MODES 4 or 5. Specific restrictions on MODE changes or Required Actions are included in the individual LCOs and discussed in the appropriate DOCs. These changes constitute editorial rewording, and presentation preferences consistent with the BWR/4 ISTS, NUREG-1433, Revision 1, and are administrative.

A13 Not Used.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.0.C requires the unit be placed in COLD SHUTDOWN (MODE 4) within 24 hours if the LCO or action requirements cannot be satisfied because of circumstances in excess of those addressed in the Specifications. ITS LCO 3.0.3 requires that the plant take action within 1 hour to initiate the shutdown, be in MODE 2 in 7 hours, be in MODE 3 in 13 hours, and be in MODE 4 in 37 hours (L1). This change requires the plant to perform the shutdown in a controlled manner which will reduce the chances for a plant transient which could challenge safety systems.

Since this change requires the plant to take action within 1 hour and to be at interim conditions, MODE 2 in 7 hours and MODE 3 in 13 hours, this change imposes additional time restraints on operations and therefore, is more restrictive. The times are consistent with NUREG-1433, Revision 1. This change has no adverse impact on safety.

M2 CTS 4.0.B does not address Frequencies specified as once. ITS SR 3.0.2 includes the phrase "For Frequencies specified as "once," the above interval extension does not apply." This is because the interval extension concept is based on scheduling flexibility for repetitive performance and these Surveillances are not repetitive in nature and essentially have no interval as measured from the previous performance. This change precludes the ability to extend these performances consistent with NUREG-1433, Revision 1. Since, CTS 4.0.B can be interpreted to apply the extension to all Surveillances, stating that the extension does not apply imposes additional requirements on operations and therefore, is more restrictive. This change has no adverse impact on safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

None

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 3.0.C requires the unit to be placed in COLD SHUTDOWN (MODE 4) within 24 hours if the LCO or action requirements cannot be satisfied because of circumstances in excess of those addressed in the Specification. ITS LCO 3.0.3 allows 37 hours to be in MODE 4 which includes the requirements (M1) to initiate the shutdown within 1 hour, be in MODE 2 within 7 hours, and be in MODE 3 in 13 hours. This change is considered less restrictive since the time to get to MODE 4 has increased by 13 hours (37 versus 24 hours). This change is acceptable since the compensating actions added in accordance with M1 and this extended time to reach MODE 4 will ensure a more continuous reduction in power and reactor coolant temperature which is within the specified maximum cooldown rate and within the capabilities of the plant. This reduces thermal stresses on components of the Reactor Coolant System and also reduces the chances for a plant transient which could challenge safety systems. This change is consistent with NUREG-1433, Revision 1.

L2 CTS 4.0.B has had the following sentence added, "If a Completion Time requires periodic performance on a "once per..." basis, the above Frequency extension applies to each performance after the initial performance." ITS SR 3.0.2 includes this statement which provides the consistency in scheduling flexibility for all performances of periodic requirements, whether they are Surveillances or Required Actions. The intent remains to perform the activity, on the average, once during each specified interval. This change is consistent with NUREG-1433, Revision 1.

L3 When it is determined that a Surveillance was not performed, CTS 4.0.C allows ACTION requirements to be delayed for up to 24 hours to permit completion of the Surveillance if the allowable outage time limits of the ACTION requirements are less than 24 hours. ITS SR 3.0.3 continues to allow a delay, from the time of discovery, up to 24 hours or up to the limit of the specified Surveillance Frequency, whichever is greater. This change is less restrictive since the delay will now apply to any Surveillance instead of those specifications with ACTION requirements of less than 24 hours. The change is also less restrictive since it will allow a delay of greater than 24 hours, but this portion of the change is described in DOC L4. The current dependance to the ACTION allowable outage time is considered not to be necessary since the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC) (continued)

L4 CTS 4.0.C allows ACTION requirements to be delayed for up to 24 hours to permit completion of the Surveillance if the allowable outage times of the ACTION requirements are less than 24 hours. ITS SR 3.0.3 allows an increase in the delay time of up to the limit of the specified Surveillance Frequency. However, a delay of greater than 24 hours is only allowed provided a risk evaluation is performed and the risk impact of delaying the Surveillance is managed. The proposed change will not allow equipment known to be inoperable to be considered Operable until the missed Surveillance is performed. If it is known that the missed Surveillance could not be met, SR 3.0.3 requires the affected LCO to be declared not met and the appropriate Condition(s) be entered. In addition, the Bases for SR 3.0.3 state that the use of the delay period is not intended to be used as an operational convenience to extend Surveillance intervals, but only for the performance of missed Surveillance(s).

This change also includes Bases details on how to implement the new allowance. The Bases provide guidance for Surveillance Frequencies that are not based on time intervals, but are based on specified plant conditions, operating conditions, or requirements of regulations. In addition, the Bases state that the licensee is expected to perform any missed Surveillance at the first reasonable opportunity, taking into account appropriate considerations, such as impact on plant risk and analysis assumptions, consideration of plant conditions, planning, availability of personnel, and the time required to perform the Surveillance. The Bases further states that the risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, Regulatory Guide (RG) 1.182, and that the missed Surveillance should be treated as an emergent condition as discussed in RG 1.182. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine that the risk increase is significant, the evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

J

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 (continued)

Therefore, since the most probable result of performing a missed Surveillance is the verification of conformance with the requirements, and the risk of extending the performance of the missed Surveillance greater than 24 hours is controlled as discussed above, this change is considered acceptable. The change is also consistent with the guidance provided in the Federal Register on September 28, 2001 (66 FR 49714), which provided the notice of availability for licensees to incorporate TSTF-358, as modified, through the Consolidated Line Item Improvement Process. J

TECHNICAL CHANGES - RELOCATIONS

None

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.0 - LCO AND SR APPLICABILITY

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

The Licensee has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change allows a more gradual plant shutdown path than allowed by CTS 3.0.C. Currently the plant has to be in Cold Shutdown within 24 hours. ITS LCO 3.0.3 requires the plant to initiate action within 1 hour to place the plant in Mode 2 (Startup/Hot Standby) within 7 hours, Mode 3 (Hot Shutdown) within 13 hours (M1) and Mode 4 (Cold Shutdown) within 37 hours. The overall time to Cold Shutdown is increased by 13 hours by the proposed change. The proposed changes will require the shutdown to proceed in a more orderly and controlled manner. This reduces thermal stresses on components of the reactor coolant system and the potential for a plant transient that could challenge safety systems under conditions to which this Specification applies. This relaxation is also acceptable based on the small probability of an event requiring the inoperable Technical Specification structures, systems and components (SSCs) to function or variables to be maintained and the desire to minimize transients. LCO 3.0.3 is only entered if the Action and Completion Time are not met and no other condition applies or if the condition of the plant is not specifically addressed by the associated actions. It is the intent of the Technical Specifications to provide action provisions, where possible, to avoid the use of LCO 3.0.3 and subsequent plant shutdown. The proposed changes to the overall shutdown Completion Times have no impact on any analyzed event. The change will not allow continuous operation when SSCs are inoperable or parameter limits are not met. In addition, the consequences of an event occurring during the proposed shutdown Completion Times are the same as the consequences of an event occurring during the existing Completion Times. The proposed change to extend the time required to reach MODE 4 is less restrictive than present provisions; however, ITS LCO 3.0.3 will provide a more orderly plant shutdown sequence without involving a significant increase in the probability or consequences of an accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE (contined)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not alter the plant configuration (no new or different type of equipment will be installed or removed) nor will the operation of the plant change. The change still ensures the plant is placed in a specified Mode or condition in a timely manner. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The relaxation in the time allowed to reach MODE 4 in accordance with proposed LCO 3.0.3 represents a relaxation over the provisions in CTS 3.0.C. However, this relaxation is acceptable based on the small probability of an event requiring the inoperable Technical Specification components to function or variables to be maintained and the desire to minimize transients. LCO 3.0.3 is only entered if the Action and Completion Time are not met and no other condition applies or if the condition of the plant is not specifically addressed by the associated ACTIONS. It is the intent of the Technical Specifications to provide action provisions, where possible, to avoid the use of LCO 3.0.3 and subsequent plant shutdown. This change will not affect a margin of safety because it has no impact on the safety analysis assumptions. The shutdown Completion Times specified in CTS 3.0.C or in ITS LCO 3.0.3 are not assumed in any analyzed accidents. This proposed change and the compensatory actions added in accordance with M1 (to initiate action within 1 hour to place the plant in MODE 2 in 7 hours and MODE 3 in 13 hours) will enhance plant safety by requiring a more orderly plant shutdown while still requiring the plant to reach MODE 4 (Cold Shutdown) within 13 hours of present provisions. Therefore, the change will not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The application of the 25% extension to Required Action Completion Times which have a specified frequency on a periodic "once per" basis has been determined to not significantly degrade the reliability that results from performing the surveillance at a specified frequency. As stated in Generic Letter 87-09, "The vast majority of surveillances do in fact demonstrate that systems or components are operable." Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   The application of the 25% extension to Required Action Completion Times which have a specified frequency on a periodic "once per" basis has been determined to not significantly degrade the reliability that results from performing the surveillance at a specified frequency. As stated in Generic Letter 87-09, "The vast majority of surveillances do in fact demonstrate that systems or components are operable." Therefore, the proposed change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS 3.0 - LCO AND SR APPLICABILITY

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

The Licensee has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   When it is determined that a Surveillance was not performed, CTS 4.0.C allows ACTION requirements to be delayed for up to 24 hours to permit completion of the Surveillance if the allowable outage time limits of the ACTION requirements are less than 24 hours. ITS SR 3.0.3 continues to allow a delay, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. Changes to the times permitted to perform a Surveillance is not considered as an initiator of any design basis accident. Therefore, this change does not significantly increase the probability of an accident previously analyzed. The most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. The added time allowance is not considered to cause the component or subsystem to become inoperable or parameter to drift out of compliance. Therefore, the consequences of an event occurring during this extended time period will be bounded by the current allowances. Therefore, this change does not significantly increase the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   When it is determined that a Surveillance was not performed, CTS 4.0.C allows ACTION requirements to be delayed for up to 24 hours to permit completion of the Surveillance if the allowable outage time limits of the ACTION requirements are less than 24 hours. ITS SR 3.0.3 continues to allow a delay, from the time of discovery, up to 24 hours or up to

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE (continued)

the limit of the specified Frequency, whichever is greater. The most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. The added time allowance is not considered to cause the component or subsystem to become inoperable or a parameter to drift out of compliance. Therefore, the consequences of an event occurring during this extended time period will be bounded by the current allowances. Therefore, this change does not result in a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 CHANGE

The Licensee has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change relaxes the time allowed to perform a missed Surveillance. The time between Surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be Operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed Surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed Surveillance will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. A missed Surveillance will not, in and of itself, introduce new failure modes or effects and any increased chance that a standby system might fail to perform its safety function due to a missed Surveillance would not, in the absence of other unrelated failures, lead to an accident beyond those previously evaluated. The addition of a requirement to assess and manage the risk introduced by the missed Surveillance will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 CHANGE (continued)

3. Does this change involve a significant reduction in a margin of safety?

The extended time allowed to perform a missed Surveillance does not result in a significant reduction in the margin of safety. As supported by the historical data, the likely outcome of any Surveillance is verification that the LCO is met. Failure to perform a Surveillance within the prescribed Frequency does not cause equipment to become inoperable. The only effect of the additional time allowed to perform a missed Surveillance on the margin of safety is the extension of the time until inoperable equipment is discovered to be inoperable by the missed Surveillance. However, given the rare occurrence of inoperable equipment, and the rare occurrence of a missed Surveillance, a missed Surveillance on inoperable equipment would be very unlikely. This must be balanced against the real risk of manipulating the plant equipment or condition to perform the missed Surveillance. In addition, parallel trains and alternate equipment are typically available to perform the safety function of the equipment not tested. Thus, there is confidence that the equipment can perform its assumed safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

[4]
[4.0]

## 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

[4.0.A]
[4.0.C]

SR 3.0.1 SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

---

[4.0.B]

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

---

[4.0.C]

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is ~~less~~ greater. This delay period is permitted to allow performance of the Surveillance. J

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. TA4

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be

A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. J

(continued)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.0 - LCO AND SR APPLICABILITY

RETENTION OF EXISTING REQUIREMENT (CLB)

None

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

PA1 Editorial changes have been made for enhanced clarity or to correct a grammatical/typographical error.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

None

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 104, 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 166, Revision 0, have been incorporated into the revised Improved Technical Specifications.

TA3 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 208, Revision 0, have been incorporated into the revised Improved Technical Specifications.

TA4 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 358, Revision 6, have been incorporated into the revised Improved Technical Specifications. J

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

None

BASES

plant PA2

LCO 3.0.3 (continued)

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

PA2 plant LCO PA3 J

The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 4 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 4, or other applicable MODE, is not reduced. For example, if MODE 2 is reached in 2 hours, then the time allowed for reaching MODE 3 is the next 11 hours, because the total time for reaching MODE 3 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, and 3, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 4 and 5 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

PA2 plant

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.~~8~~, "Spent Fuel Storage Pool Water Level." LCO 3.7.~~8~~ has an Applicability of "During movement of irradiated fuel

7 PA7

(continued)

JA2

INSERT B 3.0-4

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. 1 J

BASES

SR 3.0.3 (continued)

period of up to 24 hours or greater up to the limit of the specified Frequency, whichever is ~~less~~ applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met. TA8 PA4

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. PA2 plant

Insert SR 3.0.3-A

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

TA8

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

Insert SR 3.0.3-B

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the

(continued)

TA 8

## Insert SR 3.0.3-A

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating conditions, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

TA 8

## Insert SR 3.0.3-B

While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 71, Revision 2, have been incorporated into the revised Improved Technical Specifications.

TA2 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 8, Revision 2, have been incorporated into the revised Improved Technical Specifications. However, TSTF-8 adds a clarification to the Bases of SR 3.0.1 that allows credit to be taken for unplanned events that satisfy Surveillances. This clarification also states that this allowance also includes those SRs whose performance is precluded in a given MODE or other specified condition. This portion of the TSTF has not been adopted. As documented in the Part 9900 of the NRC Inspection Manual, Technical Guidance - Licensee Technical Specifications Interpretations, and in the ITS Bases Control Program (ITS 5.5.11), neither the Technical Specification Bases nor Licensee generated interpretations can be used to change the Technical Specification requirements. Thus, if the Technical Specifications preclude performance of an SR in certain MODES (as is the case for some SRs in ITS Section 3.8), the Bases cannot change the Technical Specifications requirement and allow the SR to be credited for being performed in the restricted MODES, even if the performance is unplanned. Therefore, only the first part of the TSTF-8 change to SR 3.0.1 has been adopted. J

TA3 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 52, Revision 3, have been incorporated into the revised Improved Technical Specifications.

TA4 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 104, Revision 0, have been incorporated into the revised Improved Technical Specifications.

TA5 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 122, Revision 0, have been incorporated into the revised Improved Technical Specifications.

TA6 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 165, Revision 0, have been incorporated into the revised Improved Technical Specifications.

TA7 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 273, Revision 2, have been incorporated into the revised Improved Technical Specifications.

TA8 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 358, Revision 6, have been incorporated into the revised Improved Technical Specifications. However, minor changes have been made for consistency. J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.0 - LCO AND SR APPLICABILITY

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

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 The paragraph in ITS LCO 3.0.4 has been moved consistent with change package BWR-26, C.1. This change was incorrectly inserted in the wrong position when NUREG-1433, Revision 1 was promulgated.

X2 ITS LCO 3.0.6 Insert B 3.0-2, provided in TSTF-71, Rev. 1, has been revised to include the sentence originally included in TSTF-71, Rev. 0. The exclusion of this sentence was identified as a typographical error and is addressed by TSTF-71, Rev. 2.

X3 ITS SR 3.0.1 has been revised to reflect the value of ≥ 800 psig consistent with ITS 3.1.4 (M6).

X4 The Bases for LCO 3.0.4 and SR 3.0.4 has been revised to reflect the possibility to enter MODE 2 from MODE 5 instead of from just MODES 3 or 4. The plant can have the Reactor Mode Switch in Refuel and complete the tensioning of all reactor vessel head closure bolts and based on Table 1.1-1 the plant will be immediately in MODE 2 without passing into MODE 3 or 4. Therefore, this modification simply corrects the Bases to reflect all possible ways of entering MODE 2.

## 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. J J

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

## 3.0 SR APPLICABILITY

| | |
|---|---|
| SR 3.0.3 (continued) | When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. |
| SR 3.0.4 | Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with Actions or that are part of a shutdown of the plant.<br><br>SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3. |

BASES

LCO 3.0.3 (continued)

of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of LCO 3.0.3 allow 37 hours for the plant to be in MODE 4 when a shutdown is required during MODE 1 operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 4, or other applicable MODE, is not reduced. For example, if MODE 2 is reached in 2 hours, then the time allowed for reaching MODE 3 is the next 11 hours, because the total time for reaching MODE 3 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed. 1 J

In MODES 1, 2, and 3, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 4 and 5 because the plant is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a plant shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the plant. An example of this is in LCO 3.7.7, "Spent Fuel Storage Pool Water Level." LCO 3.7.7 has an Applicability of "During movement of irradiated fuel assemblies in the spent fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.7 are not met while in

(continued)

# B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.10 and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the plant is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a Special Operations LCO are only applicable when the Special Operations LCO is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. I

(continued)

BASES (continued)

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met. This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of plant conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified plant conditions, operating conditions, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should

(continued)

BASES

SR 3.0.3 (continued)

include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to plant conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

(continued)

BASES

SR 3.0.4 (continued)

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the plant.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed per SR 3.0.1, which states that Surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency, on equipment that is inoperable, does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any plant shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability

(continued)

BASES

SR 3.0.4 (continued)

would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 3 from MODE 4, MODE 2 from MODE 3, 4, or 5, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, or 3. The requirements of SR 3.0.4 do not apply in MODES 4 and 5, or in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

SUMMARY OF CHANGES TO ITS SECTION 3.1 - 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 ITS markup. (The scram time to notch position 46 has been changed from 0.441 to 0.44; a comma has been added to the forth line of SR 3.1.7.6; and the wording of SR 3.1.7.11 has been modified.) | **Specification 3.1.4**<br>Retyped ITS p 3.1-14<br>**Specification 3.1.7**<br>Retyped ITS p 3.1-22 |
| NUREG ITS markup error | A minor NUREG markup error has been corrected to be consistent with the retyped ITS. (A period has been added to Required Action A.1.) | **Specification 3.1.3**<br>NUREG ITS markup p 3.1-7 |
| 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 "shuffling within the reactor pressure vessel," and the word "or" has been changed to "and" in SR 3.1.1.1; a comma has been added to a sentence in the ITS 3.1.3 Applicable Safety Analyses section; the words "THERMAL Power" have been changed to "THERMAL POWER" in SR 3.1.3.2 and SR 3.1.3.3; the apostrophe has been deleted from the ITS 3.1.4 Reference 6 word "Owners'"; the comma has been moved inside the end quote mark in ITS 3.1.5 Applicable Safety Analyses section; and the term "BWR16" has been changed to "BWR/6" in ITS 3.1.6 Reference 3.) | **Specification 3.1.1**<br>Retyped ITS Bases p B 3.1-5 and B 3.1-6<br>**Specification 3.1.3**<br>Retyped ITS Bases p B 3.1-14 and B 3.1-20<br>**Specification 3.1.4**<br>Retyped ITS Bases p B 3.1-28<br>**Specification 3.1.5**<br>Retyped ITS Bases p B 3.1-29<br>**Specification 3.1.6**<br>Retyped ITS Bases p B 3.1-38 |
| NUREG Bases markup errors | Minor NUREG markup errors have been corrected to be consistent with the retyped ITS Bases. (The word "or" has been deleted from the SR 3.1.1.1 section; the proper punctuation marks have been added to various References in ITS 3.1.1, 3.1.2, 3.1.3, and 3.1.4; and the words "of the" have been deleted from the ITS 3.1.4 LCO section.) | **Specification 3.1.1**<br>NUREG Bases markup p B 3.1-5 and B 3.1-6<br>**Specification 3.1.2**<br>NUREG Bases markup p B 3.1-12<br>**Specification 3.1.3**<br>NUREG Bases markup p B 3.1-21<br>**Specification 3.1.4**<br>NUREG Bases markup p B 3.1-23 and B 3.1-28 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Typographical errors | Minor typographical errors have been corrected in the NUREG Bases markups and retyped ITS Bases. (A comma has been deleted from the ITS 3.1.3 Background section; and the name "Lianes" has been change to "Lianas" in ITS 3.1.6 Reference 2.) | **Specification 3.1.3**<br>NUREG Bases markup p B 3.1-13<br>Retyped ITS Bases p B 3.1-13<br>**Specification 3.1.6**<br>NUREG Bases markup p B 3.1-37<br>Retyped ITS Bases p B 3.1-38 |
| Consistency issue | Minor consistency issue corrections have been made. (The revision and date of certain Bases References, which are also identified in the COLR, have been deleted and a statement that the Revision is specified in the COLR has been added. This is consistent with Specification 5.6.5.) | **Specification 3.1.1**<br>NUREG Bases markup p B 3.1-6 and B 3.1-7<br>Retyped ITS Bases p B 3.1-7<br>**Specification 3.1.4**<br>NUREG Bases markup p B 3.1-28<br>Retyped ITS Bases p B 3.1-28<br>**Specification 3.1.6**<br>NUREG Bases markup p B 3.1-37<br>Retyped ITS Bases p B 3.1-38 |
| Consistency issue | The word "solution" in the second Frequency of SR 3.1.7.9 has been changed to "piping" for consistency with the Bases. SR 3.1.7.9 requires a verification that the heat traced piping between the storage tank and the pump suction is unblocked. Thus, the Frequency should be keyed to when the piping temperature is restored, not to when the storage tank temperature is restored. The NUREG Bases states it is the piping temperature that is of concern. This change is also consistent with the most recently approved BWR ITS conversions (LaSalle and NMP2 ITS). | **Specification 3.1.7**<br>DOC M2 (DOCs p 1 of 7)<br>NUREG markup p 3.1-22<br>JFD PA1 (JFDs p 1 of 2)<br>Retyped ITS p 3.1-22 |
| Consistency issues | The word "boron" in SR 3.1.7.5, including the second Frequency, has been changed to "sodium pentaborate" for consistency with the similar words in SR 3.1.7.1, SR 3.1.7.2, SR 3.1.7.10, and SR 3.1.7.11. Also, the Bases for SR 3.1.7.9 has been modified to reflect the current manner in which the Surveillance is being performed (pumping from the storage tank and recirculating it back to the storage tank). | **Specification 3.1.7**<br>NUREG ITS markup p 3.1-21<br>JFD PA2 (JFDs p 1 of 2)<br>NUREG Bases markup p B 3.1-45<br>Retyped ITS p 3.1-21<br>Retyped ITS Bases p B 3.1-44 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Consistency issues | For consistency and clarity throughout the ITS, each of the two Scram Discharge Volumes has been described as having a header and an instrument volume. Also, the Bases for SR 3.1.8.3, describing why the Frequency is acceptable, has been modified to be consistent with the NUREG (i.e., the words "and analysis" have been deleted and the word "usually" has been added). | **Specification 3.1.8**<br><br>NUREG Bases markup p 3.1-47 and B 3.1-51<br><br>Retyped ITS Bases p B 3.1-46 and B 3.1-50 |
| Editorial | The proper References/Reference number has been provided. | **Specification 3.1.1**<br><br>NUREG Bases markup p B 3.1-5, B 3.1-6, B 3.1-7<br><br>Retyped ITS Bases p B 3.1-6 and B 3.1-7<br><br>**Specification 3.1.4**<br><br>NUREG Bases markup p B 3.1-28<br><br>Retyped ITS Bases p B 3.1-34<br><br>**Specification 3.1.6**<br><br>NUREG Bases markup p B 3.1-37<br><br>Retyped ITS Bases p B 3.1-38 |
| Editorial | The description of the plant specific reduced notch worth procedure (RNWP) has been deleted from the Applicable Safety Analyses section, since the RNWP is a subset of the BPWS. Thus the description is not necessary. A Reference has been renumbered due to this deletion. | **Specification 3.1.6**<br><br>NUREG Bases markup p B 3.1-35, Insert Page B 3.1-35 (deleted), and Insert Page B 3.1-38<br><br>JFD CLB1 (deleted) (JFDs p 1 of 2)<br><br>Retyped ITS Bases p B 3.1-35 and B 3.1-38 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Technical change | The second Frequency for SR 3.1.2.1 has been changed from "Each full power month..." to "1000 MWD/T" to be consistent with the NUREG wording. An appropriate DOC and NSHC has been provided. | **Specification 3.1.2**<br>CTS markup p 1 of 1<br>DOC L3 (DOCs p 4 of 4)<br>NSHC L3 (NSHCs p 5 of 5)<br>NUREG ITS markup p 3.1-6<br>JFD CLB1 (deleted) (JFDs p 1 of 1)<br>NUREG Bases markup p B 3.1-12<br>Bases JFD CLB1 (deleted) (Bases JFDs p 1 of 1)<br>Retyped ITS p 3.1-6<br>Retyped ITS Bases p B 3.1-12 |
| Technical change | The proper normal water pressure of the CRD accumulators has been provided. A Reference concerning this pressure has also been added. | **Specification 3.1.5**<br>NUREG Bases markup p B 3.1-33<br>JFD DB1 (JFDs p 1 of 1)<br>Retyped ITS Bases p B 3.1-33 |
| Technical change | The Discussion of Changes have been modified to properly justify deleting the requirement to replace all (i.e., both) SLC System explosive primer assemblies every 24 months. The NUREG and ITS only require firing and replacing one of the two primer assemblies every 24 months. | **Specification 3.1.7**<br>CTS markup p 2 of 5<br>DOCs M5, LA1, and L5 (DOCs p 3 of 7 and 7 of 7)<br>NUREG Bases markup p B 3.1-45<br>Retyped ITS Bases p B 3.1-43 |

BASES

PA1

PA1

Insert A from previous page

ACTIONS

E.1, E.2, E.3, E.4, and E.5 (continued)

These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

assumed to be isolated to mitigate radioactivity releases. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances as 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, SRs may need to be performed to restore the component to OPERABLE status. Action must continue until all required components are OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

PA1

Verified

PA1

Adequate SDM must be ~~demonstrated~~ to ensure that the reactor can be made subcritical from any initial operating condition. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, control rod replacement, or shuffling within the reactor pressure vessel. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 6). For the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10% Δk/k) must be added to the SDM limit of 0.28% Δk/k to account for uncertainties in the calculation.

PA1

This can be accomplished by a test (by withdrawing control rods), an evaluation, or a combination of the two.

with the highest reactivity worth control rod fully withdrawn and all other control rods fully inserted

CLB2

CLB1

X1

6

In both cases,

The SDM may be demonstrated during an in sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local criticals, where the highest worth control rod is determined by testing.

PA3

CLB1

(continued)



Revision J

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

Local critical tests require the withdrawal of out of sequence control rods. This testing would therefore require bypassing of the rod worth minimizer to allow the out of sequence withdrawal, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing—Operating") and LCO 3.10.8, "SHUTDOWN MARGIN Test — Refueling". PA1

The Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

PA1

During MODES 3 and 4, analytical calculation of SDM may be used to assure the requirements of SR 3.1.1.1 are met.

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

REFERENCES

1. UFSAR, Section 16.6 ~~10 CFR 50, Appendix A, GDC 26~~. DB1
2. UFSAR, Section ~~[15.1.38]~~. 14.6.1.2 DB2
3. NEDE-24011-P-A~~-9-US~~, "General Electric Standard Application for Reactor Fuel," Supplement for United States, Section S.2.2.3.1, ~~September 1988~~ (Revision specified in the COLR) DB3 J
4. UFSAR, Section ~~[15.1.13]~~. 14.5.4.3 DB2
5. UFSAR, Section ~~[15.1.14]~~. 14.5.4.4 DB2

PA4 PA1 7. INSERT NEXT PAGE

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

(continued)



Revision J

INSERT FROM PAGE B 3.1-6 PA1

BASES

REFERENCES (continued)

6. FSAR, Section [4.3.2.4.1].

7. 6 PA1 NEDE-24011-P-A-8, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, September 1988. DB3 J

(Revision specified in the COLR) DB3 J

BASES

ACTIONS

E.1, E.2, E.3, E.4, and E.5 (continued)

least one SGT subsystem is OPERABLE; and secondary containment isolation capability is available in each associated secondary containment penetration flow path not isolated that is assumed to isolate to mitigate radioactivity releases (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or acceptable administrative controls assure isolation capability. These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated). This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances as 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, SRs may need to be performed to restore the component to OPERABLE status. Action must continue until all required components are OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

Adequate SDM must be verified to ensure that the reactor can be made subcritical from any initial operating condition with the highest reactivity worth control rod fully withdrawn and all other control rods fully inserted. This can be accomplished by a test (by withdrawing control rods), an evaluation, or a combination of the two. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, shuffling within the reactor pressure vessel, or control rod replacement. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, "R", which is

J

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 6). 1(J)

The SDM may be demonstrated during an in-sequence control rod withdrawal or during local criticals. In both cases, the highest worth control rod is analytically determined. Local critical tests require the withdrawal of out of sequence control rods. This testing would therefore require bypassing of the rod worth minimizer to allow the out of sequence withdrawal, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing – Operating" and LCO 3.10.8, "SHUTDOWN MARGIN Test – Refueling").

The Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

During MODES 3 and 4, analytical calculation of SDM may be used to assure the requirements of SR 3.1.1.1 are met. 1(J)
During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 16.6.

2. UFSAR, Section 14.6.1.2.

3. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, Supplement for United States, Section S.2.2.3.1, (Revision specified in the COLR). J J

4. UFSAR, Section 14.5.4.3.

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

6. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, Section 3.2.4.1, (Revision specified in the COLR). J

7. UFSAR, Section 14.5.4.4. J

Specification 3.1.2



[LCO 3.1.2]

[3.1.2]

3.3 (cont'd)

restore core activity difference to within the limits within 72 hours

L1

[3.1.2]

4.3 (cont'd)

D. Reactivity Anomalies

M1

MODES 1 and 2

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1 percent Δ k. If this limit is [SR 3.1.2.1] exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken as appropriate.

[ACTION A]

A2

[LCO 3.0.4] E

[ACTION B]

See ITS 3.1.4

If Specifications 3.3.C and D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold condition within 24 hr.

L2

MODE 3 within 12 hours

M2

A1

Once within 24 hours after reaching equilibrium conditions following

M3

A3

A4

During the Startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

LA1

J

or control rod replacement

M3

1000 MWD/T during operation IN MODE 1

L3

J

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC) (continued)

L2 The CTS 3.3.E requirement to be in a cold condition within 24 hours, if CTS 3.3.D cannot be met, is being deleted. This deletion is acceptable since ITS 3.1.2 ACTION B (M2) requirement to be in MODE 3 in 12 hours, if the Required Action and associated Completion Time are not met, has been added, which places the plant in a Condition outside the ITS 3.1.2 (CTS 3.3.D) Applicability. In MODE 3 all control rods are fully inserted and therefore the reactor is in the least reactive state, where measuring core reactivity is not necessary, a continuation to cold shutdown (MODE 4) will not reduce core reactivity and therefore also is not necessary. In addition, if the reactivity anomaly specification is not met and if ITS 3.1.1 SHUTDOWN MARGIN (SDM) cannot be met, entry into the appropriate ITS 3.1.1 ACTION is required. In MODE 3, primary and secondary containment OPERABILITY is required, therefore adequate protection exists if a reactivity anomaly were to occur. The requirements of ITS 3.1.2 provide adequate protection and therefore is considered acceptable. This change is consistent with NUREG-1433, Revision 1.

L3 CTS 4.3.D requires the shutdown margin to be verified within limits every full power month (approximately 708 MWD/T). In ITS SR 3.1.2.1, the Frequency has been changed to every "1000 MWD/T during operations in MODE 1" after the first performance (the first performance requirement is described in DOC M3). Both Frequencies consider the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. The proposed change is consistent with the BWR Standard Technical Specification, NUREG-1433, Revision 1. J

TECHNICAL CHANGES - RELOCATIONS

None

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.2 - REACTIVITY ANOMALIES

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change would allow 72 hours to evaluate and determine the cause of any reactivity anomalies prior to requiring a plant shutdown. Such a reactivity anomaly is not considered an initiator of any accident previously evaluated and therefore would not affect their probability. Substantial margin exists in the analyses which predict core reactivity and in those which analyze the accidents. In addition, adequate shutdown margin is demonstrated by test during plant startup after in-vessel fuel movement or control rod replacement and is followed by a reactivity anomaly test within 24 hours of reaching equilibrium conditions at greater than 75% rated thermal power. Since the first reactivity anomaly test is typically performed within a few days following the shutdown margin demonstration the reactivity difference between the measured and predicted rod density is expected to be small. Based on experience, the reactivity differences determined by periodic performance of reactivity anomaly tests are also expected to be small, slow developing and insignificant with respect to the probability or consequences of accidents previously evaluated. Further, the consequences of an event occurring during the proposed 72 hour Completion Time are the same as the consequences of an event occurring under the current Actions. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will only provide a 72 hour Completion Time to restore the core reactivity difference to within limits before requiring a plant shutdown. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

3. Does this change involve a significant reduction in a margin of safety?

The proposed 72 hour Completion Time to restore core reactivity difference to within limits prior to requiring a plant shutdown is acceptable based on the small probability of an event occurring during this time period. Further, reactivity anomaly conditions develop slowly so there will not be a substantial change in the anomaly during the longer allowed interval before plant shutdown. Any minor decrease in the margin of safety during the additional time is offset by minimization of the potential for plant transients which may occur while shutting down the plant. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.2 - REACTIVITY ANOMALIES

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change deletes the requirement to be in a cold condition in 24 hours when the Reactivity Anomalies Specification of CTS 3.3.D is not met. Placing the plant in a cold condition does not place the plant in a less reactive condition. The reactor core is more reactive at colder temperatures, therefore the requirement to be in a cold condition does not decrease significance of the reactivity anomaly. The new requirement (M2) will be to be in MODE 3 in 12 hours (ITS 3.1.2 Required Action B.1). The proposed action is considered sufficient when Reactivity Anomalies is not met. The requirement to be in a cold condition within 24 hours if Reactivity Anomalies is not met is not considered in the initiation of any accident. Therefore, this change does not significantly increase the probability of any accident previously evaluated. The proposed ACTION exits the Applicability of the LCO and limits core reactivity. In Mode 3 the primary and secondary containment are required to be OPERABLE to limit the consequences of any design bases accident. Thus, the consequences of an accident will not be increased as a result of this change. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change will not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change limits core reactivity and other Specifications will provide additional requirements to ensure sufficient components are OPERABLE to limit any radioactivity release if an event were to occur. Therefore, this change will not create the possibility of a new or different type of accident from any accident previously analyzed.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

3. Does this change involve a significant reduction in a margin of safety?

The proposed change deletes the requirement to be in a cold condition in 24 hours when the Reactivity Anomalies Specification of CTS 3.3.D is not met. Placing the plant in a cold condition does not place the plant in a less reactive condition. The reactor core is more reactive at colder temperatures, therefore the requirement to be in a cold condition does not decrease significance of the reactivity anomaly. The new requirement (M2) will be to be in MODE 3 in 12 hours (ITS 3.1.2 Required Action B.1). The proposed ACTIONs are considered sufficient when Reactivity Anomalies is not met. The proposed action limits core reactivity and exits the Applicability of the LCO. In Mode 3 the primary and secondary containment are required to be Operable to limit the consequences of any design bases accident. Thus, the consequences of an accident will not be increased as a result of this change. Deleting this requirement to be in a cold condition when Reactivity Anomalies is not met will effectively decrease the core reactivity. This change will not impact any safety analysis assumptions. As such, no question of safety is involved. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.2 - REACTIVITY ANOMALIES

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

The Licensee has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change revises the Surveillance Frequency for the verification of the reactivity difference between the measured rod density and the predicted rod density. The proposed change continues to provide assurance that plant operation is maintained within the assumptions of the DBA and transient analysis. The proposed change in Frequency does not involve the operation of, or change to, any equipment assumed to be an initiator for any analyzed accidents. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical modification to the plant and does not introduce a new mode of operation. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The extension in the surveillance test interval is insignificant given that the proposed Frequency considers the relatively slow change in core reactivity with exposure, and operating experience related to variations in core reactivity. The proposed change does not impact the ability of the equipment to maintain the plant within acceptable limits. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

J

PA1

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.2.1 | Verify core reactivity difference between the ~~monitored~~ measured rod density and the predicted rod density is within ± 1% Δk/k. | Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement<br><br>AND<br><br>1000 MWD/T thereafter during operations in MODE 1 |

[4.3.D]
[M3]

J



Revision J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.1.2 - REACTIVITY ANOMALIES

RETENTION OF EXISTING REQUIREMENT (CLB)

None

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

PA1 Changes have been made (additions, deletions and/or changes to the NUREG) to reflect the plant specific system/structure/component nomenclature, equipment identification or description.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

None

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

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

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

PA1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

measured

rod from another core location. Also, core reactivity changes during the cycle. The 24 hour interval after reaching equilibrium conditions following a startup is based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the monitored and predicted rod density can be made. For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (no control rod movement or core flow changes) at ≥ 75% RTP have been obtained. The 1000 MWD/T Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1. J

DB1

REFERENCES

PA2

1. 10 CFR 50, Appendix A, GDC 26. UFSAR, Section 16.6

4

2. FSAR, Chapter 15. 14 DB2

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

The tests performed at this Frequency also use base data obtained during the first test of the specific cycle.



Revision J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.2 - REACTIVITY ANOMALIES

RETENTION OF EXISTING REQUIREMENT (CLB)

None

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

PA1 Editorial changes have been made for enhanced clarity or to correct a grammatical/typographical error.

PA2 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific system/structure/component nomenclature, equipment identification or description.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 JAFNPP was designed and under construction prior to the promulgation of Appendix A to 10 CFR 50 - General Design Criteria for Nuclear Power Plants. 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 FR 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. The brackets have been removed from the Reference and the proper plant specific reference included.

DB2 ITS 3.1.2 has been revised to reflect the specific JAFNPP reference requirements of, UFSAR, Chapter 14.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

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

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

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

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.2.1 | Verify core reactivity difference between the measured rod density and the predicted rod density is within ± 1% Δk/k. | Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement<br><br>AND<br><br>1000 MWD/T thereafter during operations in MODE 1 |

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

another core location. Also, core reactivity changes during the cycle. The 24 hour interval after reaching equilibrium conditions following a startup is based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the measured and predicted rod density can be made. For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (no control rod movement or core flow changes) at ≥ 75% RTP have been obtained. The 1000 MWD/T Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1. The tests performed at this Frequency also use base data obtained during the first test of the specific cycle. | J

REFERENCES

1. UFSAR, Section 16.6.

2. UFSAR, Chapter 14.

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

## 3.1 REACTIVITY CONTROL SYSTEMS

### 3.1.3 Control Rod OPERABILITY

[ITS 3.3.A.2]
[A1] LCO 3.1.3 Each control rod shall be OPERABLE.

[A1] APPLICABILITY: MODES 1 and 2.

ACTIONS

[A1] ----------NOTE----------
Separate Condition entry is allowed for each control rod.
---

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One withdrawn control rod stuck. [3.3.A.2.a] [L1] [A1] | ----------NOTE---------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow continued operation. ---------- A.1 Verify stuck control rod separation criteria met. AND | Immediately |
| | A.2 Disarm the associated control rod drive (CRD). AND | 2 hours (TA1) |
| | | (continued) |

# B 3.1 REACTIVITY CONTROL SYSTEMS

## B 3.1.3 Control Rod OPERABILITY

### BASES

**BACKGROUND**

Control rods are components of the control rod drive (CRD) System, which is the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes to ensure under conditions of normal operation, including ~~anticipated operational occurrences~~ abnormal operational transients, that specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the requirements ~~of GDC 26, GDC 27, GDC 28, and 29 (Ref. 1)~~ specified in Reference 1.

The CRD System consists of 137 locking piston ~~control rod drive mechanisms (CRDMs)~~ CRD and a hydraulic control unit for ~~each drive mechanism~~. The locking piston type CRD~~M~~ is a double acting hydraulic piston, which uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion.

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses of References 2, 3, and ~~4~~ 3.

**APPLICABLE SAFETY ANALYSES**

~~The analytical methods and assumptions used in the evaluations involving control rods are presented in References 2, 3, and 4.~~ The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical and for limiting the potential effects of reactivity insertion events caused by malfunctions in the CRD System.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.5 (continued)

affect coupling. This includes control rods inserted one notch and then returned to the "full out" position during the performance of SR 3.1.3.2. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved and operating experience related to uncoupling events.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 27, GDC 28, and GDC 29. ← UFSAR, Section 16.6. DB1 I
2. PA2 UFSAR, Section ~~[4.2.3.2.2.4]~~ 14.6
3. UFSAR, Section ~~[5A.4.3]~~ 14.5 DB2
4. ~~FSAR, Section [15.1]~~ 10 CFR 50.36(c)(2)(ii) X1
5. NEDO-21231, "Banked Position Withdrawal Sequence," Section 7.2, January 1977. PA3

## B 3.1 REACTIVITY CONTROL SYSTEMS

## B 3.1.3 Control Rod OPERABILITY

BASES

BACKGROUND

Control rods are components of the Control Rod Drive (CRD) System, which is the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes to ensure under conditions of normal operation, including abnormal operational transients, that specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the requirements specified in Reference 1.

The CRD System consists of 137 locking piston CRDs and a hydraulic control unit for each CRD. The locking piston type CRD is a double acting hydraulic piston, which uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion.

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses of References 2 and 3. 15

APPLICABLE SAFETY ANALYSES

The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical and for limiting the potential effects of reactivity insertion events caused by malfunctions in the CRD System.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The capability to insert the control rods provides assurance that the assumptions for scram reactivity in the DBA and transient analyses are not violated (Refs. 2 and 3). Since the SDM ensures the reactor will be subcritical with the highest worth control rod withdrawn (assumed single failure), the additional failure of a second control rod to insert, if required, could invalidate the demonstrated SDM and potentially limit the ability of the CRD System to hold the reactor subcritical. If the control rod is stuck at an inserted position and becomes decoupled from the CRD, a control rod drop accident (CRDA) can possibly occur. Therefore, the requirement that all control rods be OPERABLE ensures the CRD System can perform its intended function.

The control rods also protect the fuel from damage which could result in release of radioactivity. The limits protected are the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)", and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), and the fuel damage limit (see Bases for LCO 3.1.6, "Rod Pattern Control") during reactivity insertion events. J

The negative reactivity insertion (scram) provided by the CRD System provides the analytical basis for determination of plant thermal limits and provides protection against fuel damage limits during a CRDA. The Bases for LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6 discuss in more detail how the SLs are protected by the CRD System.

Control rod OPERABILITY satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

The OPERABILITY of an individual control rod is based on a combination of factors, primarily, the scram insertion times, the control rod coupling integrity, and the ability to determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram accumulator status for a control rod only affects the scram insertion times; therefore, an inoperable accumulator does not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.2 and SR 3.1.3.3 (continued)

performance and the ease of performing notch testing for fully withdrawn control rods. Partially withdrawn control rods are tested at a 31 day Frequency, based on the potential power reduction required to allow the control rod movement and considering the large testing sample of SR 3.1.3.2. Furthermore, the 31 day Frequency takes into account operating experience related to changes in CRD performance. At any time, if a control rod is immovable, a determination of the control rods OPERABILITY must be made and appropriate action taken. These SRs are modified by Notes that allows 7 days and 31 days respectively, after withdrawal of the control rod and increasing power to above the LPSP of the RWM, to perform the Surveillance. This acknowledges that the control rod must be first withdrawn and THERMAL POWER must increase to above the LPSP before performance of the Surveillance, and therefore the Notes avoid potential conflicts with SR 3.0.3 and SR 3.0.4. J

SR 3.1.3.4

Verifying that the scram time for each control rod to notch position 04 is ≤ 7 seconds provides reasonable assurance that the control rod will insert when required during a DBA or transient, thereby completing its shutdown function. This SR is performed in conjunction with the control rod scram time testing of SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," and the functional testing of SDV vent and drain valves in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlap this Surveillance to provide complete testing of the assumed safety function. The associated Frequencies are acceptable, considering the more frequent testing performed to demonstrate other aspects of control rod OPERABILITY and operating experience, which shows scram times do not significantly change over an operating cycle.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded. Above 800 psig, the scram function is designed to insert negative reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the MCPR SL, during the analyzed limiting power transient. Below 800 psig, the scram function is assumed to ~~perform during~~ mitigate the control rod drop accident (Ref. ~~8~~ 4) and, therefore, also provides protection against violating fuel damage limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis, the scram function, along with the safety/relief valves, ensure that the peak vessel pressure is maintained within the applicable ASME Code limits.

PA1 DB1

Control rod scram times satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (Ref. 5). X1

LCO

The scram times specified in Table 3.1.4-1 ~~(in the accompanying LCO)~~ are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met (Ref. 6). To account for single failures and "slow" scramming control rods, the scram times specified in Table 3.1.4-1 are faster than those assumed in the design basis analysis. The scram times have a margin that allows ~~up to approximately 7% of the~~ 10 control rods ~~(e.g., 137 x 7% = 10)~~ to have scram times exceeding the specified limits (i.e., "slow" control rods) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens ("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished

PA1 DB1 J

(continued)

BASES

REFERENCES (continued)

~~4. ESAR, Section 15.11.~~

4 ~~5.~~ NEDE-24011-P-A-9, General Electric Standard Application for Reactor Fuel, Section 3.2.4.1, September 1988, (Revision specified in the COLR) 2.2.3.1

6. Letter from R.F. Janecek (BWROG) to R.W. Starostecki (NRC), BWR Owners Group Revised Reactivity Control System Technical Specifications, BWROG-8754, September 17, 1987.

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

7. Technical Requirements Manual.

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

NOTES

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 04. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

| NOTCH POSITION | SCRAM TIMES[(a)(b)] (seconds) when REACTOR STEAM DOME PRESSURE ≥ 800 psig |
|---|---|
| 46 | 0.44 |
| 36 | 1.08 |
| 26 | 1.83 |
| 06 | 3.35 |

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when < 800 psig, are within established limits.

BASES

REFERENCES (continued)

4. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, Supplement for United States, Section S.2.2.3.1, (Revision specified in the COLR). 1 J 1 J

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

6. Letter from R.F. Janecek (BWROG) to R.W. Starostecki (NRC), BWR Owners Group Revised Reactivity Control System Technical Specifications, BWROG-8754, September 17, 1987. 1 J

7. Technical Requirements Manual.

BASES

ACTIONS

D.1 (continued)

ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods. This Required Action is modified by a Note stating that the action is not applicable if all control rods associated with the inoperable scram accumulators are fully inserted, since the function of the control rods has been performed.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

SR 3.1.5.1 requires that the accumulator pressure be checked every 7 days to ensure adequate accumulator pressure exists to provide sufficient scram force. The primary indicator of accumulator OPERABILITY is the accumulator pressure. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1100 psig (Ref. 1). Declaring the accumulator inoperable when the minimum pressure is not maintained ensures that significant degradation in scram times does not occur. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

REFERENCES

1. UFSAR, Section ~~[4.2.3.2.2.4]~~.
2. UFSAR, Section ~~[5A.4.3]~~.
3. ~~FSAR, Section [15.1].~~

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.5 - CONTROL ROD SCRAM ACCUMULATORS

RETENTION OF EXISTING REQUIREMENT (CLB)

None

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

PA1 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.

PA2 Typographical/grammatical error corrected.

PA3 The Bases have been revised for enhanced clarity.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The word "approximately" has been added for clarity. 1 J

DB2 Changes have been made to reflect the plant specific references. In addition, the brackets have been removed from the References and the plant specific References have been included.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

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

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

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

# B 3.1 REACTIVITY CONTROL SYSTEMS

## B 3.1.5 Control Rod Scram Accumulators

BASES

BACKGROUND

The control rod scram accumulators are part of the Control Rod Drive (CRD) System and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen, which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times of LCO 3.1.4, "Control Rod Scram Times."

APPLICABLE SAFETY ANALYSES

The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate (Refs. 1 and 2). OPERABILITY of each individual control rod scram accumulator, along with LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.4, ensures that the scram reactivity assumed in the DBA and transient analyses can be met. The existence of an inoperable accumulator may invalidate prior scram time measurements for the associated control rod.

The scram function of the CRD System, and therefore the OPERABILITY of the accumulators, protects the MCPR Safety Limit (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded (see Bases for LCO 3.1.4). In addition, the scram function at low reactor vessel pressure (i.e., startup conditions) provides protection against violating fuel design limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control").

Control rod scram accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

SR 3.1.5.1 requires that the accumulator pressure be checked every 7 days to ensure adequate accumulator pressure exists to provide sufficient scram force. The primary indicator of accumulator OPERABILITY is the accumulator pressure. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of approximately 1100 psig (Ref. 4). Declaring the accumulator inoperable when the minimum pressure is not maintained ensures that significant degradation in scram times does not occur. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room. J

REFERENCES

1. UFSAR, Section 14.6.

2. UFSAR, Section 14.5.

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

4. GEK-9582C, "Hydraulic Control Unit," December 1987. J

BASES

APPLICABLE SAFETY ANALYSES (continued)

Control rod patterns analyzed in Reference 1 follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to 10% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. Generic analysis of the BPWS (Ref. 1) has demonstrated that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS MODE of operation. The generic BPWS analysis (Ref. 8) also evaluates the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods.

Rod pattern control satisfies Criterion 3 of the NRC Policy Statement.

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.

APPLICABILITY

In MODES 1 and 2, when THERMAL POWER is ≤ 10% RTP, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is > 10% RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 2). In MODES 3, 4, and 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a ~~qualified member of the technical staff~~. reactor engineer PA1

Second licensed operator (Reactor Operator or Senior Reactor Operator) or PA1

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP. DB3

REFERENCES

DB2

1. NEDE-24011-P-A-9-US, General Electric Standard Application for Reactor Fuel, Supplement for United States, Section 4.2.3.1, September 1988. DB1 PA5 DB1 J (Revision specified in the COLR) J
2. "Modifications to the Requirements for Control Rod Drop Accident Mitigating System," BWR Owners Group, July 1986. PA7
3. NUREG-0979, Section 4.2.1.3.2, April 1983. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants
4. NUREG-0800, Section 15.4.9, Revision 2, July 1981. Spectrum of Rod Drop Accidents (BWR) PA7

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

Safety Evaluation Report Related to the Final Design Approval of the GESSAR II, BWR/6 Nuclear Island Design (and Supplements 1 through 5)

(continued)

Insert Ref 1 DB1


6. NEDO-10527, Rod Drop Accident Analysis For Large BWRs, March 1972.

7. NEDO-10527, Supplement 1, Rod Drop Accident Analysis For Large Boiling Water Reactors, Addendum No. 1, Multiple Enrichment Cores With Axial Gadolinium, July 1972.

8. NEDO-10527, Supplement 2, Rod Drop Accident Analysis For Large Boiling Water Reactors, Addendum No. 2, Exposed Cores, January 1973.


Insert Ref 2


12. 10 CFR 50.36(c)(2)(ii). X1


J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.6 - ROD PATTERN CONTROL

RETENTION OF EXISTING REQUIREMENT (CLB)

None

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

PA1 Changes have been made consistent with the Specification.

PA2 Changes have been made to correct a typographical error.

PA3 Not used

PA4 Not used

PA5 The quotations used in the Bases References have been removed. The Writer's Guide does not require the use of quotations.

PA6 Changes have been made to be consistent with the plant specific terminology.

PA7 The title for the Bases References have been included for clarity.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

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

DB2 Existing Reference 2 is actually an attachment to another document. The actual reference has been revised to reflect this other document in order to facilitate location of the references in the future.

DB3 The brackets have been removed and the proper plant specific value included.

BASES

APPLICABLE SAFETY ANALYSES (continued)

number of fuel rods would reach a fuel enthalpy of 170 cal/gm, which is the enthalpy limit for eventual cladding perforation.

Control rod patterns analyzed in Reference 1 follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to 10% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. Generic analysis of the BPWS (Ref. 1) has demonstrated that the 280 cal/gm fuel energy deposition limit will not be violated during a CRDA while following the BPWS mode of operation. The generic BPWS analysis (Ref. 11) also evaluates the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods.

Rod pattern control satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 12).

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.

(continued)

BASES (continued)

REFERENCES

1. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, Supplement for United States, Section S.2.2.3.1, (Revision specified in the COLR). J J

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

3. NUREG-0979, Safety Evaluation Report Related to the Final Design Approval of the GESSAR II, BWR/6 Nuclear Island Design (and Supplements 1 through 5), Section 4.2.1.3.2, April 1983. J

4. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Section 15.4.9, Spectrum of Rod Drop Accidents (BWR), Revision 2, July 1981.

5. 10 CFR 100.

6. NEDO-10527, Rod Drop Accident Analysis For Large BWRs, March 1972.

7. NEDO-10527, Supplement 1, Rod Drop Accident Analysis For Large Boiling Water Reactors, Addendum No. 1, Multiple Enrichment Cores With Axial Gadolinium, July 1972.

8. NEDO-10527, Supplement 2, Rod Drop Accident Analysis For Large Boiling Water Reactors, Addendum No. 2, Exposed Cores, January 1973.

9. NEDO-21778-A, Transient Pressure Rises Affecting Fracture Toughness Requirements For Boiling Water Reactors, December 1978.

10. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, Addenda Winter of 1966.

11. NEDO-21231, Banked Position Withdrawal Sequence, January 1977.

12. 10 CFR 50.36(c)(2)(ii). J

Specification 3.1.7

A1



Verify all heat traced piping between storage tank and pump suction is unblocked

LA1

4.4 (cont'd)

LA1

| | Item | Frequency |
|---|---|---|
| 2. [SR 3.1.7.9] | Manually initiate the system, except the explosive valves and pump solution in the recirculation path | Once per 24 Months (add second Frequency) |
| 4. M5 [SR 3.1.7.8] LA1 | Explode one of three primer assemblies manufactured in same batch to verify proper function. Then install the two remaining primer assemblies of the same batch in the explosive valves. | Once per 24 Months Staggered Test Basis L5 |
| 5. [SR 3.1.7.8] | Demineralized water shall be injected into the reactor vessel to test that valves (except explosive valves) not checked by the recirculation test are not clogged. LA1 | Once per 24 Months |
| 6. LA2 | Test that the setting of the system pressure relief valves is between 1,400 and 1,490 psig. | In accordance with the Inservice Testing Program |
| 7. | Disassemble and inspect one explosive valve so that it can be established that the valve is not clogged. Both valves shall be inspected within two test intervals. | In accordance with the Inservice Testing Program |

M2

J

A1

B. Operation with Inoperable Components

[ACTION A] From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A shall be considered fulfilled, and continued operation permitted, provided that:

1. The component is returned to an operable condition within 7 days.

add ACTION B L2

B. Operation with Inoperable Components

When a component becomes inoperable its redundant component shall be verified to be operable immediately and daily thereafter.

L3

DISCUSSION OF CHANGES
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) 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 3.4.C states that the solution temperature including the pump suction piping temperature has to be maintained above the temperature limits. CTS 4.4.C.2 requires that the solution temperature be checked at least once per day. The ITS has two separate surveillances (SR 3.1.7.2 and SR 3.1.7.3) which require that the temperature of the sodium pentaborate solution (SR 3.1.7.2) and the temperature of the pump suction piping (SR 3.1.7.3) be verified every 24 hours. Since the pump suction piping temperature has always been a requirement for SLC OPERABILITY, having a separate SR to verify this temperature is considered an administrative change. This is consistent with NUREG-1433, Revision 1.

## TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 4.4.A.1 requires the verification that all valves (manual, power operated, or automatic) in the system flowpath that is not locked, sealed or otherwise secured in position is in the correct position. There are no power operated or automatic valves in the system except for the explosive valves. This Surveillance is included as ITS SR 3.1.7.6 for all manual valves, and a new requirement has been added to verify the continuity of each explosive charge (ITS SR 3.1.7.4). Since the ITS is more explicit on the method of verification for the explosive valve this change is considered more restrictive on plant operation. This change is consistent with NUREG-1433, Revision 1.

M2 CTS 4.4.A.3 requires verification that heat traced piping between the SLC storage tank and the pump suction is unblocked (by manually initiating the system, except explosive valves, and pump boron solution from the SLC storage tank through the recirculation path) once every 24 months. ITS SR 3.1.7.9 requires verification that heat traced piping between the SLC storage tank and the pump suction is unblocked once per 24 months and "Once within 24 hours after piping temperature is restored within the limits of Figure 3.1.7-2." The addition of this second Surveillance Frequency represents a more restrictive change necessary to ensure the piping is unblocked after conditions have existed with the

TECHNICAL CHANGES - MORE RESTRICTIVE

M2 (continued)

potential for causing the piping to become blocked due to precipitation of boron from solution.

M3 CTS 4.4.C.1 has requirements for checking the concentration of sodium pentaborate in the SLC Tank after certain events which could affect boron concentration occur (adding water to tank, adding boron to tank, or if temperature of solution in tank drops below the temperature limit). The CTS does not specify any time requirement for performing these checks. ITS (SR 3.1.7.5) adds a time limit of 24 hours into the requirement to check sodium pentaborate concentration after additions to the SLC Tank are made (water or boron). This ensures that the concentration is checked on a timely basis after additions to the tank are made rather than the current open ended specification. SR 3.1.7.5 also adds a second time requirement to check the concentration within 24 hours after solution temperature is restored within limits. This checks for the amount of boron that may have precipitated out of solution. The addition of new requirements reflects a more restrictive change necessary to ensure SLC System Operability is adequately maintained.

M4 CTS 4.4.C.4 requires that the enrichment of the Boron-10 (in the SLC tank) be checked once per 24 months, but the CTS contains no requirement for checking the Boron-10 enrichment of sodium pentaborate being added to the tank. ITS SR 3.1.7.10 requires that a Boron-10 enrichment verification be done prior to adding sodium pentaborate to the tank. Since the enrichment of a batch/lot of sodium pentaborate will not change with time, a single isotopic test of any given batch/lot can suffice as the required analysis for any number of mixings and additions from that batch/lot. For sodium pentaborate supplied and purchased under controls assuring appropriate 10CFR50, Appendix B and ANSI N45.2 compliance, the required analysis may be satisfied by certified vendor analytical test results. While this is consistent with current practice, this SR is considered more restrictive in that the requirement is not expressly stated in the CTS.

Once the Boron-10 is in the SLC tank the enrichment of the solution will not change. ITS SR 3.1.7.5 requires that the concentration of the boron solution in the SLC tank be verified within 24 hours after the boron addition. ITS SRs 3.1.7.1, 3.1.7.2 and 3.1.7.3 verify proper boron solution volume and temperature. ITS SR 3.1.7.11 (retained from CTS 4.4.C.4) verifies the enrichment of boron in the SLC tank every 24 months. These verifications, in addition to proposed ITS SR 3.1.7.10, help maintain the required quantity of B-10 in the tank. ITS SR 3.1.7.10 is

TECHNICAL CHANGES - MORE RESTRICTIVE

M4 (continued)

considered more restrictive but provides assurance that SLC System Operability is adequately maintained. This change is consistent with NUREG-1433, Revision 1.

M5 CTS 4.4.A.4 requires the firing of SLC primer assemblies prior to being installed into the plant, but does not require firing of the primer assemblies during the flow testing of CTS 4.4.A.5. ITS SR 3.1.7.8 will require the firing of an installed primer assembly as part of the SLC flow test to ensure the primer assembly opens the associated SLC valve. This requirement is considered more restrictive but provides assurance that SLC System Operability is adequately maintained. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The details of method of performing CTS 4.4.A.2 to verify flow by recirculating demineralized water to the test tank; the details in CTS 4.4.A.3, to demonstrate all piping between the SLC storage tank and the pump suction is unblocked (by manually initiating the system, except the explosive valves and pump solution in the recirculation path); the details in CTS 4.4.A.5 to verify flow through the SLC subsystem into the reactor pressure vessel (to test that the valves except explosive valves not checked by the recirculation test are not clogged); and the details in CTS 4.4.A.4 to explode one of three primer assemblies manufactured in same batch to verify proper function and (as modified by DOC L5) that the replacement primer assembly is from a batch previously tested are proposed to be relocated to the Bases. These details are not necessary to ensure that the SLC System is maintained Operable. The requirements of ITS 3.1.7 and SRs 3.1.7.7, 3.1.7.8, and 3.1.7.9 are adequate to ensure the capability to provide flow through each SLC subsystem to the test tank and into the reactor pressure vessel, to ensure the piping between the SLC storage tank and the pump suction is unblocked, and to ensure SLC System Operability. Therefore, the relocated details are not necessary 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.

LA2 The testing requirements of CTS 4.4.A.6 (to verify the proper operation and setpoints of the relief valves) and CTS 4.4.A.7 (to disassemble and inspect one explosive valve) are proposed to be relocated to the Inservice Testing (IST) Program. These testing requirements demonstrate the SLC System relief valves and explosive valves are OPERABLE.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA2 (continued)

However, the IST Program, required by 10 CFR 50.55a, provides requirements for the testing of all ASME Code Class 1, 2, and 3 valves in accordance with Section XI of the ASME Code. ITS Section 5.5.7 provides controls over the IST Program. These controls are adequate to ensure the required testing to demonstrate Operability is performed. Therefore, the relocated requirements are not necessary to be in the ITS to provide adequate protection of the public health and safety. Changes to the relocated requirements in the IST Program will be controlled by the provisions of 10 CFR 50.59.

LA3 CTS 3.4.C contains detailed information concerning the boron solution for the SLC storage tank, and what support components and variables are required to assure SLC OPERABILITY is maintained. The ITS relocates this detailed information to the Bases for Specification 3.1.7. The requirements of ITS 3.1.7 including the LCO, ACTIONS and Surveillances are adequate to ensure SLC System OPERABILITY. Therefore, the relocated details are not necessary 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.

LA4 The detail in CTS Figure 3.4-2 that the saturation temperature of enriched sodium pentaborate solution curve includes a 10°F margin is proposed to be relocated to the Bases. The requirements in ITS SR 3.1.7.2 to verify the temperature of sodium pentaborate solution is within the limits of Figure 3.1.7-2 (CTS Figure 3.4-2) and Figure 3.1.7-2 (Sodium Pentaborate Solution Temperature Versus Concentration Requirements curve) are adequate to ensure the proper evaluation is performed and therefore help ensure SLC System OPERABILITY. Therefore, the relocated details are not necessary 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.

LB1 The requirements in CTS 4.4.C.3 to calibrate the temperature and level elements is proposed to be relocated to the Technical Requirements Manual (TRM). These temperature and level indications do not necessarily relate directly to SLC System OPERABILITY. In general NUREG-1433, Revision 1, does not specify requirements for equipment which only provide indication to support OPERABILITY of a system or component. Control of the availability of, and necessary compensatory activities if not available, for indications, monitoring instruments, and alarms are addressed by plant operational procedures and policies. Therefore, the SLCs temperature and level instrument surveillances are

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LB1 (continued)

removed from the Technical Specifications and relocated to the TRM. The requirements in ITS 3.1.7 including the LCO, ACTIONS and Surveillances are adequate to ensure the SLC System is Operable. Therefore, the relocated requirements are not necessary to be in the ITS to provide adequate protection of the public health and safety. At ITS implementation, the relocated requirement will be incorporated by reference into the UFSAR. As such changes to the relocated requirements in the TRM will be controlled by the provisions of 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS requires that the Standby Liquid Control System be Operable during a period when fuel is in the reactor and prior to startup from cold condition. This System need not be Operable when the reactor is in the cold condition, control rods are fully inserted and CTS 3.3.A (Reactivity Limitations) is met. The ITS 3.1.7 Applicability is MODES 1 and 2. The current Applicability corresponds to MODES 1, 2 and may even imply MODES 3, 4 and 5 with any control rod withdrawn. This change is less restrictive since the new Applicability does not include MODES 3, 4, and 5. The SLC system is not needed during Hot or Cold Shutdown (MODES 3 or 4) since control rods can only be withdrawn in accordance with Section 3.10, "Special Operations," and adequate SDM prevents criticality under these conditions. While in the refueling MODE, the SLC System is not needed because only a single control rod can be withdrawn and adequate SDM prevents criticality when under these conditions.

L2 CTS 3.4.C includes an action to restore certain components (e.g., tank heaters) or variables (e.g., sodium pentaborate volume-concentration and temperature requirements) within 8 hours or take action to be in hot shutdown in the next 12 hours. All the components or variables discussed in CTS 3.4.C will cause both subsystems of the SLC System to be inoperable. However, the list is not all inclusive of the possible events which could lead to both subsystems being inoperable. ITS 3.1.7, ACTION B is being added to allow the entire SLC System (e.g., both pumps) to be inoperable for any reason up to 8 hours prior to requiring a plant shutdown. The 8 hours provides time to restore minor problems (e.g., some pump inoperabilities) prior to requiring a plant shutdown. The 8 hours is considered acceptable since the time is short, the SLC System is not the primary method of shutting down the plant, reduces the possibility of plant shutdown transients and the probability of an ATWS event is very small.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CTS 4.4.B requires that when a SLC subsystem or component becomes inoperable, the redundant subsystem or component be verified to be OPERABLE immediately and daily thereafter. ITS 3.1.7 does not have this cross system check. This change will allow credit to be taken for normal periodic Surveillances as a verification of OPERABILITY and availability of the remaining SLC subsystem. The periodic Frequencies specified to verify OPERABILITY of the remaining SLC subsystem has been shown to be adequate to ensure equipment OPERABILITY. As stated in NRC Generic Letter 87-09, "It is overly conservative to assume that systems or components are inoperable when a surveillance requirement has not been performed. The opposite is in fact the case; the vast majority of surveillances demonstrate the systems or components in fact are operable." Therefore, reliance on the specified Surveillance intervals does not result in a reduced level of confidence concerning the equipment availability. The ITS and current BWR operating philosophy accept the philosophy of system OPERABILITY based on satisfactory performance of monthly, quarterly, refueling interval, post-maintenance or other specified performance tests without requiring additional testing when another system is inoperable (except for diesel generator testing, which is not being changed).

L4 CTS 4.4.A.1 requires that each SLC subsystem "valve (manual, power operated, or automatic) in the system flow path that is not locked, sealed or otherwise secured in position, is in the correct position" once per 31 days. ITS SR 3.1.7.6 requires that "each SLC subsystem manual valve (see M1) 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" every 31 days. The proposed change permits the SLC subsystem to be considered OPERABLE as long as the valves can be manually realigned to their correct position. The Bases stipulates that this realignment must be capable of being done from the control room, or locally by a dedicated operator at the valve control. The SLC System is a manually initiated system. Therefore allowing the system to be considered OPERABLE whenever the system valves can be correctly aligned does not reduce the level of safety and is considered acceptable. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L5 CTS 4.4.A.5 requires that every 24 months demineralized water be injected into the reactor vessel to test that valves (except explosive valves) not checked by the recirculation test (CTS 4.4.A.3) are not clogged. This test involves testing entire subsystems; including portions common to both subsystems as well as non-common portions. As such, testing either subsystem can satisfy the necessary testing for the common portions of both subsystems. To accomplish this, ITS SR 3.1.7.8 requires the verification of flow through one SLC subsystem from the pump into reactor pressure vessel every 24 months on a STAGGERED TEST BASIS (i.e., such that the subsystems use for the test are alternated each 24 months). Since the CTS could be inferred to require testing both subsystems each 24 months, this change is a relaxation in the frequency of testing an individual subsystem (i.e., on the Staggered Test Basis), and is classified as a less restrictive change. In addition, CTS 4.4.A.4 requires replacing both primer assemblies every 24 months. ITS SR 3.1.7.8 will only require replacing one primer assembly every 24 months; the one that has been fired during performance of the SR. Over the course of two surveillance intervals (48 months), both primer assemblies will be required to be replaced. Testing of the non-common portions, which are also the subject of the relaxed testing frequency, are appropriately surveilled by other ITS SRs: specifically, each pump is tested per the IST Program as required by SR 3.1.7.7, the continuity of each explosive charge is verified every 31 days in accordance with SR 3.1.7.4, the temperature of pump suction piping is verified within limits every 24 hours with SR 3.1.7.3, and proper manual valve position is verified every 31 days as required by SR 3.1.7.6. These surveillance tests, which are performed more frequently than the proposed surveillance interval of SR 3.1.7.8, provide assurance that unacceptable conditions associated with the SLC System will be detected in a timely manner. This change is also consistent with NUREG-1433, Revision 1. J

TECHNICAL CHANGES - RELOCATIONS

None

SURVEILLANCE REQUIREMENTS

| | SURVEILLANCE | | FREQUENCY |
|---|---|---|---|
| [4.4.c.2] | SR 3.1.7.1 | Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1~~, or ≥ [4530] gallons~~. DB2 | 24 hours |
| [4.4.c.2] | ~~SR~~ 3.1.7.2 | Verify temperature of sodium pentaborate solution is within the limits of Figure 3.1.7-2. DB2 | 24 hours DB2 |
| [4.4.c.2] | ~~SR~~ 3.1.7.3 | Verify temperature of pump suction piping is within the limits of Figure 3.1.7-2. | 24 hours |
| [M1] | SR 3.1.7.4 | Verify continuity of explosive charge. | 31 days |
| [4.4 C.1] [M3] | SR 3.1.7.5 | Verify the concentration of ~~boron~~ sodium pentaborate in solution is within the limits of Figure 3.1.7-1. DB2 PA 2 | 31 days<br>AND<br>Once within 24 hours after water or ~~boron~~ is added to solution<br>AND<br>Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2. DB2 |

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.7.6 [4.4.A.1][M1][L4] | Verify each SLC 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. DB3 | 31 days |
| SR 3.1.7.7 [4.4.A.2] | Verify each pump develops a flow rate ≥ [~~41.2~~] 50 gpm at a discharge pressure ≥ [~~1190~~] 1275 psig. DB2 | In accordance with the Inservice Testing Program ~~or 92 days~~ CLB1 |
| [4.4.A.4] SR 3.1.7.8 [4.4.A.5][M5] | Verify flow through one SLC subsystem from pump into reactor pressure vessel. | [~~18~~] 24 months on a STAGGERED TEST BASIS CLB2 |
| SR 3.1.7.9 [4.4.A.3] [M2] | Verify all heat traced piping between storage tank and pump suction is unblocked. | [~~18~~] 24 months<br>AND<br>Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2 PA1 Piping CLB2 CLB2 J |
| SR 3.1.7.10 [4.4.C.4] [M4] | Verify sodium pentaborate enrichment is ≥ [~~60.0~~] 34.7 atom percent B-10. DB2 | Prior to addition to SLC tank DB5 |

[4.4.C.4] INSERT SR 3.1.7.11 CLB3



REVISION ~~D~~ J

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The brackets have been removed and the Frequency "In accordance with the Inservice Testing Program" retained consistent with the current licensing basis in CTS 4.4.A.2.

CLB2 The brackets have been removed and a 24 month Frequency included in SR 3.1.7.8 and SR 3.1.7.9 (first frequency) consistent with CTS 4.4.A.4 and CTS 4.4.A.3, respectively. In addition, the brackets have been removed from SR 3.1.7.9 and the second frequency has been added in accordance with M2.

CLB3 The requirements of CTS 4.4.C.4 for verifying Boron-10 enrichment of the sodium pentaborate solution in the SLC tank on a 24 month Frequency are retained as SR 3.1.7.11. Although the provisions of SR 3.1.7.10 are adequate to ensure proper Boron-10 enrichment, periodic verification of the SLC solution enrichment is a good practice providing added assurance that the proper Boron-10 enrichment is maintained.

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

PA1 The second Frequency for ISTS SR 3.1.7.9 (ITS SR 3.1.7.9) is being changed from being based on solution temperature to piping temperature. The SR requires a verification that all heat traced piping is unblocked. A change in solution temperature in the tank does not necessarily have an impact on the piping temperature, as long as the piping heat trace circuit is functioning properly. The intent of the second Frequency is to ensure that, if the heat tracing is inoperable such that piping temperature falls below the limits of Figure 3.1.7-2, after the heat tracing is restored to OPERABLE status and the piping temperature is within the limits of Figure 3.1.7-2 the piping is still unblocked. This is supported by the ISTS Bases description for this second Frequency, which describes the requirement as required to be performed after "piping" temperature is restored. J

PA2 The proper JAFNPP nomenclature has been used. This is also consistent with SR 3.1.7.1 and SR 3.1.7.2. J

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 NUREG-1433 ACTION A, is not applicable to JAFNPP and, has been deleted. JAFNPP requires the same concentration of boron in solution to meet the original licensing basis of the SLC System (cold shutdown) as it does for the ATWS rule (10 CFR 50.62). Therefore low boron concentration

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 (continued)

would result in both SLC subsystems being inoperable. The remaining Conditions and Required Actions have been renumbered or revised to reflect this deletion.

DB2 The brackets have been removed and the proper limits included.

DB3 The brackets have been removed and the information deleted since the system does not include any power operated or automatic valves other than the explosive valves.

DB4 NUREG Figures 3.1.7-1 and Figure 3.1.7-2 have been modified in accordance with the current requirements.

DB5 The brackets have been removed and SR 3.1.7.10 retained in accordance with CTS 4.4.C.4 and DOC M4.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

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

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.7 (continued)

requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is [in accordance with the Inservice Testing Program or 92 days].

SR 3.1.7.8 and SR 3.1.7.9

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 36 months at alternating 18 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all heat traced piping between the boron solution storage tank and the suction inlet to the injection pumps is unblocked ensures that there is a functioning flow path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping is unblocked is to pump from the storage tank to the test tank.

(continued)



REVISION ~~D~~ J

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.7.1 | Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1. | 24 hours |
| SR 3.1.7.2 | Verify temperature of sodium pentaborate solution is within the limits of Figure 3.1.7-2. | 24 hours |
| SR 3.1.7.3 | Verify temperature of pump suction piping is within the limits of Figure 3.1.7-2. | 24 hours |
| SR 3.1.7.4 | Verify continuity of explosive charge. | 31 days |
| SR 3.1.7.5 | Verify the concentration of sodium pentaborate in solution is within the limits of Figure 3.1.7-1. | 31 days<br>AND<br>Once within 24 hours after water or sodium pentaborate is added to solution<br>AND<br>Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2 |

J

J

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.7.6 | Verify each SLC subsystem manual 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 |
| SR 3.1.7.7 | Verify each pump develops a flow rate ≥ 50 gpm at a discharge pressure ≥ 1275 psig. | In accordance with the Inservice Testing Program |
| SR 3.1.7.8 | Verify flow through one SLC subsystem from pump into reactor pressure vessel. | 24 months on a STAGGERED TEST BASIS |
| SR 3.1.7.9 | Verify all heat traced piping between storage tank and pump suction is unblocked. | 24 months<br>AND<br>Once within 24 hours after piping temperature is restored within the limits of Figure 3.1.7-2 |
| SR 3.1.7.10 | Verify sodium pentaborate enrichment is ≥ 34.7 atom percent B-10. | Prior to addition to SLC tank |
| SR 3.1.7.11 | Verify sodium pentaborate enrichment in solution in the SLC tank is ≥ 34.7 atom percent B-10. | 24 months |

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.5

This Surveillance requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron in the storage tank is maintained per Figure 3.1.7-1. SR 3.1.7.5 must be performed anytime boron or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. SR 3.1.7.5 must also be performed anytime the temperature is restored to within the limits of Figure 3.1.7-2, to ensure that no significant boron precipitation occurred. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.7

Demonstrating that each SLC System pump develops a flow rate ≥ 50 gpm at a discharge pressure ≥ 1275 psig by recirculating demineralized water to the test tank ensures that pump performance has not degraded during the surveillance interval. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms pump and motor capability and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

SR 3.1.7.8 and SR 3.1.7.9

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve primer assembly. The replacement primer assembly for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.8 and SR 3.1.7.9 (continued)

valve pathway tested should be alternated such that both complete flow paths are tested every 48 months at alternating 24 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all heat traced piping between the boron solution storage tank and the suction inlet to the injection pumps is unblocked ensures that there is a functioning flow path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping is unblocked is to manually initiate the system, except the explosive valves, and pump from the storage tank and recirculating it back to the storage tank. Upon completion of this verification, the pump suction piping must be flushed with demineralized water to ensure piping between the storage tank and pump suction is unblocked. 10 J

The 24 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. This is especially true in light of the temperature verification of this piping required by SR 3.1.7.3. However, if, in performing SR 3.1.7.3, it is determined that the temperature of this piping has fallen below the specified minimum, SR 3.1.7.9 must be performed once within 24 hours after the piping temperature is restored to within the limits of Figure 3.1.7-2.

SR 3.1.7.10

Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. Isotopic

(continued)

## B 3.1 REACTIVITY CONTROL SYSTEMS

## B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

### BASES

**BACKGROUND**

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line with two valves in series. Each header is connected to a common vent line with two valves in series for a total of four vent valves. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

**APPLICABLE SAFETY ANALYSES**

The Design Basis Accident and transient analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100 (Ref. 2); and

b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR 100 (Ref. 2), and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.3 (continued)

30 60 seconds after receipt of a scram signal is based on the bounding leakage case evaluated in the accident analysis (Ref. 2). Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3 overlap this Surveillance to provide complete testing of the assumed safety function. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

DB1 3 LLB2 24 LLB2 24 1 J

REFERENCES

1. PA4 UFSAR, Section 4.2.3.2.2.3 3.5.5.2 DB2
2. 10 CFR 100.
3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981. PA5
4. 10 CFR 50.36(c)(2)(ii). X1

1 J



Revision J

## B 3.1 REACTIVITY CONTROL SYSTEMS

## B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (each SDV consisting of a header and an instrument volume), each receiving approximately one half of the control rod drive (CRD) discharges. Each instrument volume has a drain line each having two valves in series for a total of four drain valves. Each header is connected to a separate vent line each having two valves in series for a total of four vent valves. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

J

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100 (Ref. 2); and

b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR 100 (Ref. 2), and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves allow

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.3 (continued)

(Ref. 3). Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3 overlap this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint. 10 J

REFERENCES

1. UFSAR, Section 3.5.5.2.

2. 10 CFR 100.

3. NUREG-0803, Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping, August 1981.

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

SUMMARY OF CHANGES TO ITS SECTION 3.2 - REVISION J

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Retyped ITS typographical error | A minor typographical error in the retyped ITS has been corrected. (The acronym "(FRTP)" has been deleted, consistent with the NUREG markup.) | **Specification 3.2.4**<br><br>Retyped ITS p 3.2-5 |
| Retyped Bases typographical errors | Minor typographical errors in the retyped ITS have been corrected. (The words "Design Basis" have been changed to "design basis" in the ITS 3.2.1 Background section; the words "of 0.84" has been deleted from the ITS 3.2.1 Applicable Safety Analyses section; a comma has been moved inside the end quote mark in the ITS 3.2.4 Background section; and the comma has been moved inside the end quote mark in two places in ITS 3.2.4 Applicable Safety Analyses section.) | **Specification 3.2.1**<br><br>Retyped Bases p B 3.2-1 and B 3.2-2<br><br>**Specification 3.2.4**<br><br>Retyped Bases p B 3.2-13 and B 3.2-14 |
| NUREG Bases markup errors | Minor NUREG markup errors have been corrected to be consistent with the retyped ITS Bases. (The word "Safety" has been added to ITS 3.2.2 Reference 8; the proper Reference number has been added to the ITS 3.2.4 Background section; and the words "Allowable value" have been changed to "Allowable Value" in the ITS 3.2.4 LCO Section.) | **Specification 3.2.2**<br><br>NUREG Bases markup p B 3.2-10<br><br>**Specification 3.2.4**<br><br>NUREG Bases markup p B 3.2-15 and B 3.2-17 |
| Typographical error | The dash has been removed from between the words "Flow" and "Biased" in the Background section. | **Specification 3.2.4**<br><br>NUREG Bases markup p B 3.2-14<br><br>Retyped ITS Bases p B 3.2-13 |
| Consistency issue | Minor consistency issue corrections have been made. (The revision and date of certain Bases References, which are also identified in the COLR, have been deleted and a statement that the Revision is specified in the COLR has been added. This is consistent with Specification 5.6.5.) | **Specification 3.2.1**<br><br>NUREG Bases markup p B 3.2-4<br><br>Retyped ITS Bases p B 3.2-4<br><br>**Specification 3.2.2**<br><br>NUREG Bases markup p B 3.2-9<br><br>Retyped ITS Bases p B 3.2-8<br><br>**Specification 3.2.4**<br><br>NUREG Bases markup p Insert Page B 3.2-19<br><br>Retyped ITS Bases p B 3.2-18 |
| Consistency issue | The word "or" has been deleted from LCO 3.2.4.a, and the word "Function" has been added after the words "(Flow Biased)" in all places that where the term "APRM Neutron Flux - High (Flow Biased) Allowable Value" is used. | **Specification 3.2.4**<br><br>NUREG markup p 3.2-5 and 3.2-6<br><br>Retyped ITS p 3.2-5 and 3.2-6 |

SUMMARY OF CHANGES TO ITS SECTION 3.2 - REVISION J

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Editorial | The proper References have been provided for ITS 3.2.1 and ITS 3.2.2 Bases. | **Specification 3.2.1**<br><br>NUREG Bases markup p B 3.2-2, B 3.2-4, and B 3.2-5<br><br>Retyped Bases p B 3.2-2 and B 3.2-4<br><br>**Specification 3.2.2**<br><br>NUREG Bases markup p B 3.2-6, B 3.2-7, and B 3.2-10<br><br>Retyped Bases p B 3.2-5, B 3.2-6, and B 3.2-9 |

BASES

APPLICABLE SAFETY ANALYSES (continued)

which turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and AOOs. A complete discussion of the analysis code is provided in Reference 9. DB1

LOCA analyses are then performed to ensure that the above determined APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference ~~10~~ 8. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR. DB1

a conservative multiplier is applied to the exposure dependent APLHGR limits for two loop operation (Ref. 5 and 7.) DB3

For single recirculation loop operation, the MAPFAC multiplier is limited to a maximum of 0.75 (Ref. 5). This maximum limit is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA. DB1 J

The APLHGR satisfies Criterion 2 of the NRC Policy Statement. 10 CFR 50.36(c)(2)(ii) (Ref. 9) X1 J

LCO

The APLHGR limits and specified in the COLR are the result of the fuel design, DBA and transient analyses. For two recirculation loops operating, the limit is determined by multiplying the smaller of the $MAPFAC_p$ and $MAPFAC_f$ factors times the exposure dependent APLHGR limits. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," PA2 DB1

for each lattice type as a function of average planar exposure and is approved by the NRC DB2

(continued)



Revision J

BASES

ACTIONS

B.1 (continued)

operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on ~~both engineering judgment and~~ the recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDO-24011-P-A "General Electric Standard Application for Reactor Fuel" ~~(latest approved version)~~. (Revision specified in the COLR)
2. FSAR, Chapter ~~[5]~~ 3.
3. FSAR, Chapter ~~[6]~~.
4. FSAR, Chapter ~~[15]~~ 14.
5. ~~[Plant specific single loop operation]~~.
6. ~~[Plant specific load line limit analysis]~~.
7. ~~[Plant Specific Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvements (ARTS) Program]~~.
8. ~~NEDO-30130-A, "Steady State Nuclear Methods," May 1985~~.

Supplemental Reload Licensing Report for James A. FitzPatrick (Revision specified in the COLR)

NEDO-24243, General Electric Boiling Water Reactor Load Line Limit Analysis for James A. FitzPatrick Nuclear Power Plant, February 1980

NEDC-32016P-1, Power Uprate Safety Analysis For James A. FitzPatrick Nuclear Power Plant, April 1993, including Errata and Addenda Sheet No. 1, dated January 1994.

(continued)



Revision J

BASES

REFERENCES (continued)

9. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978. DB3

10. [Plant specific loss of coolant accident analysis].

8. NEDC-31317P Revision 2, James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis, April 1993. DB3 1 J

9. 10 CFR 50.36 (c)(2)(ii). X1 1 J

J



Revision J

# B 3.2 POWER DISTRIBUTION LIMITS

## B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during abnormal operational transients and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46. J

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), abnormal operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, 4, 5, 6, 7, and 8.

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during abnormal operational transients for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure to ensure adherence to fuel design limits during the limiting abnormal operational transients (Refs. 5, 6, and 7).

LOCA analyses are then performed to ensure that the above determined APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference 8. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

For single recirculation loop operation, a conservative multiplier is applied to the exposure dependent APLHGR limits for two loop operation (Refs. 5 and 7). This maximum limit is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA. J

The APLHGR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 9). J

LCO

The APLHGR limits specified in the COLR are the result of the fuel design, and DBA and transient analyses. For two recirculation loops operating, the limit is determined for each lattice type as a function of average planar exposure and is approved by the NRC. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent APLHGR limit by a conservative multiplier determined by a specific single recirculation loop analysis (Ref. 5).

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and analyses of LOCAs and transients that are assumed to occur at high power levels. Design calculations and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

(continued)

BASES (continued)

REFERENCES

1. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, (Revision specified in the COLR). J

2. UFSAR, Chapter 3.

3. UFSAR, Chapter 6.

4. UFSAR, Chapter 14.

5. Supplemental Reload Licensing Report for James A. FitzPatrick (Revision specified in the COLR). J

6. NEDO-24243, General Electric Boiling Water Reactor Load Line Limit Analysis For James A. FitzPatrick Nuclear Power Plant, February 1980.

7. NEDC-32016P-1, Power Uprate Safety Analysis For James A. FitzPatrick Nuclear Power Plant, April 1993, including Errata and Addenda Sheet No. 1, dated January 1994. J

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

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

## B 3.2 POWER DISTRIBUTION LIMITS

## B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

PA2

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) 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). The operating limit MCPR is established to ensure that no fuel damage results during ~~anticipated operational occurrences~~ (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

PA1

abnormal operational transients

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

DB2

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 6, 7, and 8. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

DB1

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

INSERT ASA

states (MCPR$_f$ and MCPR$_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 6, 7, and 8). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 9) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power dependent MCPR limits (MCPR$_p$) are determined mainly by the one dimensional transient code (Ref. 10). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow MCPR$_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the MCPR$_f$ and MCPR$_p$ limits.

The MCPR values for each fuel assembly must remain above the operating limit MCPR.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as

actual values for key plant parameters

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of $\tau$, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4,"Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter $\tau$ must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1 and SR 3.1.4.2 because the effective scram speed distribution may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in $\tau$ expected during the fuel cycle.

or after maintenance that could affect scram times

TA1

TA1

SR 3.1.4.4

PA2

Fuel Rod Failure as a Consequence of Departure From Nucleate Boiling or Dry Out,

DB2

REFERENCES

1. NUREG-0562, June 1979.
2. NEDO-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).
3. FSAR, Chapter [4]. 3
4. FSAR, Chapter [6].
5. FSAR, Chapter [15]. 14
6. (Plant specific single loop operation).
7. (Plant specific load line limit analysis).

PA1

4

NEDO-24281, Fitz Patrick Nuclear Power Plant Single-Loop Operation, August 1980.

NEDO-24243, General Electric Boiling Water Reactor Load Line Limit Analysis for James A. FitzPatrick Nuclear Power Plant, February 1980

all references except as indicated

(Revision specified in the COLR)

5

(continued)



REVISION ~~0~~ 1

BASES

REFERENCES (continued)

8. [Plant specific Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvements (ARTS) Program].

9. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

10. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

DB1

DB2

NEDC-32016P-1, Power Uprate Safety Analysis For James A FitzPatrick Nuclear Power Plant, April 1993, Including Errata and Addenda Sheet No. 1, dated January 1994.

J

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

X1

DB2

9. Supplemental Reload Licensing Report for James A. FitzPatrick (Revision specified in the COLR).

J

## B 3.2 POWER DISTRIBUTION LIMITS

## B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) 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). The operating limit MCPR is established to ensure that no fuel damage results during abnormal operational transients. Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the abnormal operational transients to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 6, 7, 8, and 9. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and core exposure to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 6, 7, 8, and 9). A generator load reject without bypass and a feedwater controller transient normally result in the worst case MCPR transients for a given fuel cycle. During operations at low core flows the MCPR operating limit must be increased by a factor of $K_f$ (specified in the COLR) which is derived from the recirculation flow runout transient and is a function of core flow. This will ensure the MCPR safety limit is not exceeded during a recirculation flow runout event. J

The MCPR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 10). J

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is a function of exposure, control rod scram times and core flow. The MCPR values for each fuel assembly must remain above the operating limit MCPR.

APPLICABILITY

The MCPR operating limits are primarily derived from the analyses of transients that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of actual values for key plant parameters important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 25% RTP. This trend is expected to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of $\tau$, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4,"Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter $\tau$ must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in $\tau$ expected during the fuel cycle.

REFERENCES

1. NUREG-0562, Fuel Rod Failure as a Consequence of Departure From Nucleate Boiling or Dry Out, June 1979.

2. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, (Revision specified in the COLR). J

3. UFSAR, Chapter 3.

4. UFSAR, Chapter 6.

5. UFSAR, Chapter 14.

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

(continued)

BASES

REFERENCES (continued)

7. NEDO-24243, General Electric Boiling Water Reactor Load Line Limit Analysis For James A. FitzPatrick Nuclear Power Plant, February 1980.

8. NEDC-32016P-1, Power Uprate Safety Analysis For James A. FitzPatrick Nuclear Power Plant, April 1993, including Errata and Addenda Sheet No. 1, dated January 1994. J

9. Supplemental Reload Licensing Report for James A. FitzPatrick (Revision specified in the COLR). J

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

PA2 CLB1

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints (Optional)

[L1] LCO 3.2.4

a. MFLPD shall be less than or equal to Fraction of RTP; PA2 [J]

b. Each required APRM ~~setpoint~~ specified in the COLR shall be made applicable; or

c. Each required APRM gain shall be adjusted ~~such that the APRM readings are > 100% times MFLPD~~ as specified in the COLR PA1

[4.1.B] APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| [L1] A. Requirements of the LCO not met. | A.1 Satisfy the requirements of the LCO. | 6 hours |
| [L1] B. Required Action and associated Completion Time not met. | B.1 Reduce THERMAL POWER to < 25% RTP. | 4 hours |

PA1

Neutron Flux - High (Flow Biased) Function Allowable Value [J]

PA3 LLB1

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.4.1 | ----------NOTE---------- Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4 ~~Item~~ b or c requirements. ---------- | PA2 |
| [4.1.B] | Verify MFLPD is within limits. LCO 3.2.4. | Once within 12 hours after ≥ 25% RTP<br><br>AND<br><br>24 hours thereafter |
| SR 3.2.4.2 | ----------NOTE---------- Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4 ~~Item~~ a requirements. PA2 ---------- | |
| [4.1.B] | Verify ~~APRM setpoints~~ or gains are adjusted for the calculated MFLPD. | 12 hours |

APRM Neutron Flux-High (Flow Biased) Function Allowable Value J

PA1



REVISION ~~B~~ J

PA1 LB1

## B 3.2 POWER DISTRIBUTION LIMITS

### B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint~~s~~ ~~(Optional)~~

**BASES**

**BACKGROUND**

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design," GDC 13, "Instrumentation and Control," GDC 20, "Protection System Functions," and GDC 23, "Protection against Anticipated Operation Occurrences" (Ref. 1). This LCO is provided to require the APRM gain or APRM ~~flow biased scram setpoints~~ to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

DB1

Applicable design criteria is discussed in UFSAR, Section 16.6

The condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP), where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

Neutron Flux-High (Flow Biased) Function Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b)

PA2

$$\frac{\text{MFLPD}}{\text{FRTP}} > 1,$$

indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by a gain adjustment on the APRMs or adjustment of the APRM ~~setpoints~~. Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP and thus maintains RTP margins for APLHGR ~~and~~ MCPR.

Neutron Flux-High (Flow Biased) Function Allowable Value

and LHGR

DB4

3

J

Neutron Flux-High (Flow Biased) Function Allowable Value

PA2

The normally selected APRM ~~setpoints~~ position the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The ~~setpoints are~~ flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow,

PA2

Allowable Value is

(continued)

BASES

**BACKGROUND (continued)**

the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM setpoints are supported by the analyses presented in References 1 and 2 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR and MCPR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the flow biased APRM scram setpoints may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.

The APRM neutron flux signal is also adjusted to more closely follow the fuel cladding heat flux during power transients. The APRM neutron flux signal is a measure of the core thermal power during steady state operation. During power transients, the APRM signal leads the actual core thermal power response because of the fuel thermal time constant. Therefore, on power increase transients, the APRM signal provides a conservatively high measure of core thermal power. By passing the APRM signal through an electronic filter with a time constant less than, but approximately equal to, that of the fuel thermal time constant, an APRM transient response that more closely follows actual fuel cladding heat flux is obtained, while a conservative margin is maintained. The delayed response of the filtered APRM signal allows the flow biased APRM scram levels to be positioned closer to the upper bound of the normal power and flow range, without unnecessarily causing reactor scrams during short duration neutron flux spikes. These spikes can be caused by insignificant transients such as performance of main steam line valve surveillances or momentary flow increases of only several percent.

**APPLICABLE SAFETY ANALYSES**

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR and MCPR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

FSAR safety analyses (Refs. 2 and 3) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR and MCPR) occurs.

(continued)



Revision J

PA2

BASES

PA4 %

LCO (continued)

c. Increasing APRM gains to cause the APRM to read greater than 100 times MFLPD (in %). This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM flow biased scram setpoints are reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM flow biased scram setpoints. Adjusting APRM gain or setpoints is equivalent to MFLPD less than or equal to FRTP, as stated in the LCO.

PA2

Neutron Flux - High (Flow Biased) Function Allowable Value

is

modifying the

is

maintaining

is

For compliance with LCO Item b (APRM setpoint adjustment) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," are required to be adjusted. In addition, each APRM may be allowed to have its gain or setpoints adjusted independently of other APRMs that are having their gain or setpoints adjusted.

modification

Function 2.b

Allowable Value

is

or modified

Allowable Value

modified or

APPLICABILITY

The MFLPD limit, APRM gain adjustment, and APRM flow biased scram and associated setdowns are provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the reactor is operating at ≥ 25% RTP.

modification is

PA2

ACTIONS

A.1

PA2

is

If the APRM gain or setpoints are not within limits while the MFLPD has exceeded FRTP, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit

(continued)



Revision J

DB23 K1

Insert Ref


3. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, (Revision specified in the COLR). 15

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

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

6. GENE-637-004-0295, 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.

7. 10 CFR 50.36(c)(2)(ii).

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

LCO 3.2.4

a. MFLPD shall be less than or equal to Fraction of RTP;

b. Each required APRM Neutron Flux–High (Flow Biased) Function Allowable Value specified in the COLR shall be made applicable; or

c. Each required APRM gain shall be adjusted as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. Requirements of the LCO not met. | A.1 Satisfy the requirements of the LCO. | 6 hours |
| B. Required Action and associated Completion Time not met. | B.1 Reduce THERMAL POWER to < 25% RTP. | 4 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.4.1 | ----------NOTE----------<br>Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4.b or LCO 3.2.4.c requirements.<br>----------<br><br>Verify MFLPD is within limits. | Once within 12 hours after ≥ 25% RTP<br><br>AND<br><br>24 hours thereafter |
| SR 3.2.4.2 | ----------NOTE----------<br>Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4.a requirements.<br>----------<br><br>Verify required APRM Neutron Flux-High (Flow Biased) Function Allowable Value or ARPM gains are adjusted for the calculated MFLPD. | 12 hours |

## B 3.2 POWER DISTRIBUTION LIMITS

## B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable design criteria is discussed in UFSAR, Section 16.6 (Ref. 1). This LCO is provided to require the APRM gain or APRM Neutron Flux-High (Flow Biased) Function Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b) to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit. 15

The condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP), where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

$$\frac{\text{MFLPD}}{\text{FRTP}} > 1,$$

indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by a gain adjustment on the APRMs or adjustment of the APRM Neutron Flux–High (Flow Biased) Function Allowable Value. Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP and thus maintains RTP margins for APLHGR, MCPR, and LHGR.

The normally selected APRM Neutron Flux–High (Flow Biased) Function Allowable Value positions the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The Allowable Value is flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core 17

(continued)

BASES

BACKGROUND (continued)

flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow, the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM Allowable Value is supported by the analyses presented in References 2 and 3 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR, MCPR, and LHGR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the APRM Neutron Flux–High (Flow Biased) Function Allowable Value may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution. In addition, the APRM Neutron Flux–High (Flow Biased) Function provides protection from reactor thermal hydraulic instability consistent with Boiling Water Reactors Owners' Group Long-Term Solution, Option I-D (Refs. 4, 5 and 6).

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR, MCPR, and LHGR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

The safety analyses (Refs. 2 and 3) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR, MCPR, and LHGR) occurs. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," and LCO 3.2.3, "Linear Heat Generation Rate (LHGR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of either the APLHGR, the MCPR, or the LHGR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients

J

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

circuitry. SR 3.2.4.1 and SR 3.2.4.2 have been modified by Notes which clarify that the respective SR does not have to be met if the alternate requirement demonstrated by the other SR is satisfied. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1) and LHGR (LCO 3.2.3). The 24 hour Frequency is based on the recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

The 12 hour Frequency of SR 3.2.4.2 requires a more frequent verification than if MFLPD is less than or equal to FRTP. When MFLPD is greater than FRTP, more rapid changes in power distribution are typically expected.

REFERENCES

1. UFSAR, Section 16.6.

2. UFSAR, Section 14.5.

3. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, (Revision specified in the COLR). J

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

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

6. 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.

7. 10 CFR 50.36(c)(2)(ii).

SUMMARY OF CHANGES TO ITS SECTION 3.3 - 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 SR numbers for Table 3.3.1.1-1 Function 2.a have been changed from SR 3.3.1.1.5 and SR 3.3.1.1.6 to SR 3.3.1.1.6 and SR 3.3.1.1.7; the word "NOTES" has been changed to "NOTE" in SR 3.3.1.2.4; the word "Channel" has been changed to "channel" in ITS 3.3.5.1 Required Action C.2; the Frequency "92 day" has been changed to "92 days" in SR 3.3.6.2.3; and the word "vacuum" has been changed to "air removal" in ITS 3.3.7.2 Surveillance Requirements Note.) | **Specification 3.3.1.1**<br>Retyped ITS p 3.3-6<br>**Specification 3.3.1.2**<br>Retyped ITS p 3.3-13<br>**Specification 3.3.5.1**<br>Retyped ITS p 3.3-33<br>**Specification 3.3.6.2**<br>Retyped ITS p 3.3-60<br>**Specification 3.3.7.2**<br>Retyped ITS p 3.3-65 |
| NUREG ITS markup errors | Minor NUREG markup errors have been corrected to be consistent with the retyped ITS. (The inequality sign ">" has been changed to "≤" in the Allowable Value of Table 3.3.5.1-1 Function 1.d (shown in INSERT Function 1.d); the Table 3.3.6.1-1 Function 2.f Applicability has been changed from MODES 1, 2, and 3 to MODES 1[b] and 2[b]; the inequality sign "≤" has been changed to ">" in the Allowable Value of Table 3.3.6.1-1 Function 2.e; the inequality sign "≤" has been added to the Allowable Value for Table 3.3.6.1-1 Function 5.b; the acronym "RWC" has been changed to "RWCU" in Table 3.3.6.1-1 Note d; the comma has been deleted from the Function Title for Table 3.3.6.1-1 Function 6.b; the bracket has been removed from Table 3.3.6.2-1 Note b; the ITS 3.3.7.1 Applicability has been changed from "MODES 1, 2, AND 3" to MODES 1, 2, and 3 and the word "the" has been added to the second Applicability; the word "Note" has been changed to "NOTE" in ITS 3.3.7.2 Required Action A.2 and SR 3.3.7.2.2; and the word "NOTES" has been changed to "NOTE" in the ITS 3.3.8.1 Surveillance Requirements.) | **Specification 3.3.5.1**<br>NUREG ITS markup p Insert page 3.3-42<br>**Specification 3.3.6.1**<br>NUREG ITS markup p Insert page 3.3-58, 3.3-61, and 3.3-62<br>**Specification 3.3.6.2**<br>NUREG ITS markup p 3.3-66<br>**Specification 3.3.7.1**<br>NUREG ITS markup p Insert page 3.3-71<br>**Specification 3.3.7.2**<br>NUREG ITS markup p Insert page 3.3-74a and Insert page 3.3-74c<br>**Specification 3.3.8.1**<br>NUREG ITS markup p 3.3-76 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| 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 parentheses around the words "EHC Oil Pressure-Low have been deleted and a comma added after the term "B1" in the Background section of ITS 3.3.1.1; the word "result" is changed to "results" in the ASA, LCO, and Applicability section of ITS 3.3.1.1 Function 1.a; two paragraphs have been combined into one paragraph in the ASA, LCO, and Applicability section of ITS 3.3.1.1 Function 2.a; a comma and a period is added to the header of the ASA, LCO, and Applicability section of ITS 3.3.1.1, Functions 7.a and 7.b; the term "RPS" has been added to the first sentence of SR 3.3.1.1.4; the word "Test" has been changed to "TEST" in the next to last sentence of the first paragraph of SR 3.3.1.1.8/11; the word "sensory" has been changed to "sensing" in SR 3.3.1.1.10; the word "assoicated" has been changed to "associated" in SR 3.3.1.2.5 and SR 3.3.1.2.6; the words "To", "And", and "For" have been decapitalized and the dash after the word "To" has been deleted in ITS 3.3.2.1 Reference 8; the word "while" has been deleted in SR 3.3.2.2.2; the word "To" and "And" have been decapitalized in ITS 3.3.2.2 Reference 4; the word "primaryindication" has been changed to "primary indication" in the ITS 3.3.3.1 LCO section for Function 1; the words "Hydrogen Analyzer" and "Oxygen Analyzer" have been decapitalized in the ITS 3.3.3.1 LCO section for Function 8; the word "associatedindicator" has been changed to "associated indicator" in ITS 3.3.3.1 LCO section for Function 11; the word "additionalfailure" has been changed to "additional failure" in the ITS 3.3.3.1 Actions section; the word "the" has been added to SR 3.3.3.1.2 and SR 3.3.3.1.3; the word "generater" has been changed to "generator" and the term "Level 3" has been changed to "Level 2" in ITS 3.3.4.1 Background section; the word "satifies" has been changed to "satisfies" and the word "uncertanties" has been changed to "uncertainties" in ITS 3.3.4.1 ASA, LCO, and Applicability section; the word "number" has been changed to "numbers" in ITS 3.3.4.1 Actions A.1 and A.2; the word "channelSurveillance" has been changed to "channel Surveillance" in ITS 3.3.4.1 Surveillance Requirements;<br>CONTINUED ON NEXT PAGE | **Specification 3.3.1.1**<br><br>Retyped ITS Bases p B 3.3-3, B 3.3-4, B 3.3-7, B 3.3-10, B 3.3-18, B 3.3-29, B 3.3-31, and B 3.3-33<br><br>**Specification 3.3.1.2**<br><br>Retyped ITS Bases p B 3.3-46<br><br>**Specification 3.3.2.1**<br><br>Retyped ITS Bases p B 3.3-59<br><br>**Specification 3.3.2.2**<br><br>Retyped ITS Bases p B 3.3-66 and B 3.3-67<br><br>**Specification 3.3.3.1**<br><br>Retyped ITS Bases p B 3.3-70, B 3.3-73, B 3.3-75, B 3.3-76, and B 3.3-79<br><br>**Specification 3.3.4.1**<br><br>Retyped ITS Bases p B 3.3-86, B 3.3-87, B 3.3-88, B 3.3-91, and B 3.3-93 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Retyped ITS Bases typographical errors (continued) | the word "component" has been changed to "pump and valve" (three places), the word "permissive" has been added (two places), the word "pool" has been added, the word "additional" has been deleted, the word "the" has been added, the word "handswitches" has been changed to "hand switches", the word "System" has been changed to "system" (two places), and the word "the" has been deleted in the ITS 3.3.5.1 Background section; the words "as appropriate" have been added, a paragraph has been split into two paragraphs, the word "(Level2)" has been changed to "(Level 2)", two paragraphs have been combined into one paragraph, a period has been changed to a comma, the word "ANAYLYSES" has been changed to "ANALYSES,", the letter "D" has been changed to "C", and the letter "C" has been changed to "D" in the ITS 3.3.5.1 ASA, LCO, and Applicability section; two paragraphs have been combined into one paragraph in ITS 3.3.5.1 Actions F.1 and F.2 and SR 3.3.5.1.1; a period has been added after the number 2.a in the Header for ITS 3.3.5.1 Functions 1.a, 2.a; the word "on" has been changed to "upon" in SR 3.3.5.1.5; dashes have been added between the words "Loss of Coolant" in ITS 3.3.5.1 Reference 4; the word "To" has been decapitalized in ITS 3.3.5.2 Reference 3; the word "vessel" has been added, a comma has been deleted, parentheses has been placed around the term "Level 3", and the word "water" has been added to the ITS 3.3.6.1 Background section; one paragraph has been split into two paragraphs (three places), the word "or" has been changed to "on", the number "3" has been changed to "2", the word "suchthat" has been changed to "such that", the words "to be" have been added, the word "This" has been changed to "The", the words "reactor pressure vessel" have been changed to "RPV", the words "inside the" have been added, the word "Primary" has been decapitalized, two paragraphs have been combined into one paragraph, the word "and" has been deleted (two places), and an extra sentence related to the Allowable Value has been deleted in the ITS 3.3.6.1 ASA, LCO, and Applicability section; periods have been deleted (the period at the end of the first Function) from the Functions 2.a and 2.g and Functions 2.b and 2.d Headers in the ITS 3.3.6.1 ASA, LCO, and Applicability Bases section; the word "continous" has been changed to "continuous" and a comma has been deleted in the ITS 3.3.6.1 Actions section; a comma has been added and two paragraphs have been combined into one paragraph in the ITS 3.3.6.1 Surveillance Requirements; the words "For", "To", and "And" have been decapitalized in ITS 3.3.6.1 Reference 8; the word "of" has been changed to "to" in ITS 3.3.6.1 Reference 11; the word "trip" changed to "nominal", the word "nominal" added, and the word ", respectively" deleted in the ITS 3.3.6.2 ASA, LCO, and Applicability section; the word "automatic" and a comma have been deleted in the ITS 3.3.6.2 Actions section; two paragraphs have been combined into one paragraph and the word "underthese" has been changed to "under these" in the ITS 3.3.6.2 Surveillance Requirements; the word "Specification" has been changed to "Specifications" and the words "For", "To", and "And" have been decapitalized in the ITS 3.3.6.2 Reference 7;<br>CONTINUED ON NEXT PAGE | **Specification 3.3.5.1**<br><br>Retyped ITS Bases p B 3.3-97, B 3.3-98, B 3.3-100, B 3.3-101, B 3.3-102, B 3.3-103, B 3.3-105, B 3.3-106, B 3.3-107, B 3.3-114, B 3.3-116, B 3.3-119, B 3.3-120, B 3.3-122, B 3.3-130, B 3.3-133, B 3.3-134, and B 3.3-135<br><br>**Specification 3.3.5.2**<br><br>Retyped ITS Bases p B 3.3-148<br><br>**Specification 3.3.6.1**<br><br>Retyped ITS Bases p B 3.3-153, B 3.3-156, B 3.3-157, B 3.3-161, B 3.3-162, B 3.3-165, B 3.3-166, B 3.3-167, B 3.3-168, B 3.3-173, B 3.3-174, B 3.3-179, B 3.3-183, and B 3.3-184<br><br>**Specification 3.3.6.2**<br><br>Retyped ITS Bases p B 3.3-187, B 3.3-191, B 3.3-192, B 3.3-193, B 3.3-194, B 3.3-196, and B 3.3-197 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Retyped ITS Bases typographical errors (continued) | the words "as appropriate" have been added to the ITS 3.3.7.1 ASA, LCO and Applicability section; the words "line steam" have been changed to "steam line" in the ITS 3.3.7.2 Background section; the word "does" has been changed to "doses" in the ITS 3.3.7.2 ASA section; the word "system" has been changed to "systems" in the ITS 3.3.7.2 Actions C.1, C.2, and C.3 section; the words "limits are" have been changed to "limit is" in the ITS 3.3.7.3 LCO section; a comma has been added to the ITS 3.3.8.1 Background section; and the word "the" has been added and the word "4.16kV" has been changed to "4.16 kV".) | **Specification 3.3.7.1**<br><br>Retyped ITS Bases p B 3.3-199<br><br>**Specification 3.3.7.2**<br><br>Retyped ITS Bases p B 3.3-203 and B 3.3-207<br><br>**Specification 3.3.7.3**<br><br>Retyped ITS Bases p B 3.3-211<br><br>**Specification 3.3.8.1**<br><br>Retyped ITS Bases p B 3.3-216, B 3.3-217, and B 3.3-220 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| NUREG Bases markup errors | Minor NUREG Bases markup errors have been corrected to be consistent with the retyped ITS Bases. (A comma has been added to the ITS 3.3.1.1 Background section; a comma has been added to SR 3.3.1.1.10; the word "acutal" has been changed to "actual" in INSERT SR 3.3.1.1.15-1; the words "Supplement 1" are moved to just after the actual Reference document number in ITS 3.3.1.1 Reference 6; a period has been added to ITS 3.3.1.1 Reference 17; the second dash is deleted in ITS 3.3.1.1 Reference 18; the second dash has been deleted and the words "Supplement 1" added in ITS 3.3.2.1 Reference 9; the range of the hydrogen analyzer has been added into Insert LCO-8; the proper date of Regulatory Guide 1.97, Revision 3 has been provided (i.e., 1983 vice 1985) in ITS 3.3.3.1 Reference 1; a period has been added after the initial "C" in ITS 3.3.3.1 Reference 2; the words "Pressure High-trip" have been changed to "Pressure-High trip and the words "Reactor High Pressure-High" have been changed to Reactor Pressure-High" in ITS 3.3.4.1 INSERT BKGD; the word "meets" has been changed to "satisfies" in the ITS 3.3.4.1 ASA, LCO, and Applicability section; the word "serveillance" has been changed to "surveillance" in ITS 3.3.4.1 INSERT ASA; the proper document number has been provided and the word "Evalution" has been changed to "Evaluation" in ITS 3.3.4.1 INSERT REF; a comma has been added and the word "and" has been deleted in ITS 3.3.5.1 HPCI Background section; the word "References" has been changed to "Reference" (two places), a comma has been added, "(Level) 1" has been changed to "(Level 1)", a comma has been deleted, the word "tip" has been changed to "trip", the words "sensing high pump discharge pressure" have been added, and the word "ensough" has been changed to "enough" in the ITS 3.3.5.1 ASA, LCO and Applicability section; the words "this Function provides" have been changed to "these Functions provide" in ITS 3.3.5.1 Actions B.1, B.2, and B.3; a comma has been added to ITS 3.3.5.1 Actions C.1 and C.2; the word "an" has been changed to "a" in SR 3.3.5.1.3; the second dash has been deleted and the word "BWR" has been added to ITS 3.3.5.1 Reference 7; "(Ref. 2)" has been changed to "(Ref. 3)" in ITS 3.3.5.2 Actions C.1; a period has been added to ITS 3.3.5.2 Reference 2; a dash has been added to the document number for ITS 3.3.5.2 Reference 3; (CONTINUED ON NEXT PAGE) | **Specification 3.3.1.1**<br><br>NUREG Bases markup p B 3.3-1, B 3.3-30, Insert Page B 3.3-32, and B 3.3-33<br><br>**Specification 3.3.2.1**<br><br>NUREG Bases markup p B 3.3-55<br><br>**Specification 3.3.3.1**<br><br>NUREG Bases markup p B Insert Page 3.3-68, B 3.3-73, and Insert Page B 3.3-73<br><br>**Specification 3.3.4.1**<br><br>NUREG Bases markup p Insert Page B 3.3-91, B 3.3-92, Insert Page B 3.3-92, and Insert Page B 3.3-100<br><br>**Specification 3.3.5.1**<br><br>NUREG Bases markup p B 3.3-103, B 3.3-104, B 3.3-108, B 3.3-109, B 3.3-112, B 3.3-115, Insert Page B 3.3-115, B 3.3-118, Insert Page B 3.3-119, B 3.3-126, B 3.3-127, B 3.3-137, and B 3.3-138<br><br>**Specification 3.3.5.2**<br><br>NUREG Bases markup p B 3.3-146 and B 3.3-151 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| NUREG Bases markup errors (continued) | a comma has been added and the word "However," has been added to the ITS 3.3.6.1 Background section; a period has been added (five places), the word "This" has been changed to "The", the word "Footnote" changed to "footnote", a period added to the end of the Function number in the Function Header in INSERT Function 1.f, a close parenthesis has been deleted and a close quotation mark has been added in INSERT Function 1.f, the words of INSERT Function 2.a(2) have been modified to reflect the retyped ITS (related to the description of the reference point for the low water level Allowable Value), the word "pressure" has been added, the word "form" changed to "from", the words "design basis" capitalized, a comma has been deleted, and periods have been added to the end of the two Function numbers in the Function Headers in INSERT Function 7 in the ITS 3.3.6.1 ASA, LCO, and Applicability section; a comma has been added and the word "subsystem(s)" has been changed to "subsystem" in ITS the 3.3.6.1 Actions section; parentheses have been added around the word "Note" in the ITS 3.3.6.1 Surveillance Requirements; two commas have been added and the word "Function" has been changed to "Functions" in ITS 3.3.6.1 INSERT SR Note; a period has been added to ITS 3.3.6.1 Reference 4; a comma has been added to ITS 3.3.6.1 Reference 8; the word "Requirements" has been added to ITS 3.3.6.1 Reference 10; a comma has been added to the ITS 3.3.6.2 Background section; the words "worst case" have been added to ITS 3.3.6.2 INSERT ASA; a period has been added to ITS 3.3.6.2 Reference 3; the word "Sections" has been changed to "Section" and the end bracket deleted in ITS 3.3.6.2 Reference 5; a close parenthesis has been added to ITS 3.3.6.2 Reference 8; a comma has been added to the ITS 3.3.7.1 ASA, LCO, and Applicability section; the words "worst case" have been added to the second line and the words ". The Allowable Values are then derived from the trip setpoints by accounting for" have been added to the sixth line of ITS 3.3.7.1 INSERT ASA; the word "vacuum" has been deleted and the word "limits" changed to "limit" in the ITS 3.3.7.2 LCO section; a period has been added to ITS 3.3.7.2 Reference 4; the words "high enough" have been added and the word "analytical" has been changed to "analytic" in the ITS 3.3.7.3 LCO section; a comma has been deleted and a period has been added to the ITS 3.3.8.1 ASA, LCO, and Applicability section; and the word "Electrical" has been changed to "Electric" and a period has been added to ITS 3.3.8.2 Reference 3.) | **Specification 3.3.6.1**<br><br>NUREG Bases markup p B 3.3-152, B 3.3-155, B 3.3-159, B 3.3-160, Insert Page B 3.3-161, B 3.3-162, Insert Page B 3.3-162, B 3.3-163, Insert Page B 3.3-164a, Insert Page B 3.3-164b, B 3.3-165, Insert Page B 3.3-174a, Insert Page B 3.3-174b, B 3.3-176, B 3.3-179, B 3.3-180, Insert Page B 3.3-180, B 3.3-184, and Insert Page B 3.3-184<br><br>**Specification 3.3.6.2**<br><br>NUREG Bases markup p B 3.3-185, Insert Page B 3.3-186, B 3.3-196, and B 3.3-197<br><br>**Specification 3.3.7.1**<br><br>NUREG Bases markup p B 3.3-208 and Insert Page B 3.3-208<br><br>**Specification 3.3.7.2**<br><br>NUREG Bases markup p Insert Page B 3.3-219b and Insert Page B 3.3-219k<br><br>**Specification 3.3.7.3**<br><br>NUREG Bases markup p Insert Page B 3.3-219k and Insert Page B 3.3-219l<br><br>**Specification 3.3.8.1**<br><br>NUREG Bases markup p B 3.3-220 and B 3.3-222<br><br>**Specification 3.3.8.2**<br><br>NUREG Bases markup p B 3.3-233 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Typographical errors | Minor typographical errors have been corrected in the Discussion of Changes, NUREG ITS markups, NUREG Bases markups, the retyped ITS, and the retyped ITS Bases. (A comma has been added to the ITS 3.3.1.1 Background section; a comma has been added to the ASA, LCO, and Applicability section of ITS 3.3.1.1; a dash has been added between the words "Flux" and "High" in the ASA, LCO, and Applicability section of ITS 3.3.1.1 Function 2.b; a comma has been deleted in the ASA, LCO, and Applicability section of ITS 3.3.1.1 Function 2.c; a comma has been added to the ASA, LCO, and Applicability section of ITS 3.3.1.1 Function 10; a comma has been added to SR 3.3.1.1.2; the word "contacts(s)" has been changed to "contact(s)" in SR 3.3.1.1.3 and SR 3.3.1.1.8 Bases; the word "reponse" has been changed to "response" in SR 3.3.1.1.15; the word "contacts(s)" has been changed to "contact(s)" in SR 3.3.1.2.5 Bases; the sign ">" has been changed to "≥" in ITS 3.3.2.1 DOC M4; the term "Average Planar Linear Heat Generation Rate" in the ITS 3.3.2.2 Applicability Bases section has been capitalized since it is a definition; the word "contacts(s)" has been changed to "contact(s)" in SR 3.3.2.1.1, SR 3.3.2.1.2, and SR 3.3.2.1.7 Bases; the word "contacts(s)" has been changed to "contact(s)" in SR 3.3.2.2.2 Bases; the value "10 E8 R/hr" has been changed to "1E8 R/hr" in the ITS 3.3.3.1 LCO Bases for Function 5; the document number has been changed from "6681" to "688" in ITS 3.3.3.1 Bases Reference 5; the word "the" has been deleted in one location from the second paragraph of the INSERT BKGD for the ITS 3.3.4.1 NUREG Bases markup and in two locations from the fourth paragraph of the retyped ITS 3.3.4.1 Background Bases; Table "3.3.5.1" has been changed to "3.3.5.1-1", Table "3.3.5.2" has been changed to "3.3.5.2-1" and a semicolon has been added after the words "Function 1" in the ASA, LCO, and Applicability Bases section for ITS 3.3.4.1 Function a; the word "contacts(s)" has been changed to "contact(s)" in SR 3.3.4.1.2 Bases; CONTINUED ON NEXT PAGE | **Specification 3.3.1.1**<br><br>NUREG ITS markup p 3.3-3<br><br>NUREG Bases markup p B 3.3-1, B 3.3-3, Insert Page B 3.3-9, B 3.3-11, B 3.3-19, Insert Page B 3.3-27, Insert Page B 3.3-29, and Insert Page B 3.3-32<br><br>Retyped ITS p 3.3-3<br><br>Retyped ITS Bases p B 3.3-5, B 3.3-12, B 3.3-13, B 3.3-21, B 3.3-28, B 3.3-31, and B 3.3-35<br><br>**Specification 3.3.1.2**<br><br>NUREG Bases markup p Insert Page B 3.3-41<br><br>Retyped ITS Bases p B 3.3-45<br><br>**Specification 3.3.2.1**<br><br>DOC M4 (DOCs p 3 of 9)<br><br>NUREG Bases markup p Insert Page B 3.3-51 and Insert Page B 3.3-53<br><br>Retyped ITS Bases p B 3.3-55 and B 3.3-57<br><br>**Specification 3.3.2.2**<br><br>NUREG Bases markup p B 3.3-58 and Insert Page B 3.3-61<br><br>Retyped ITS Bases p B 3.3-62 and B 3.3-65<br><br>**Specification 3.3.3.1**<br><br>NUREG Bases markup p Insert Page B 3.6-67 and Insert Page B 3.3-73<br><br>Retyped ITS Bases p B 3.3-72 and B 3.3-80<br><br>**Specification 3.3.4.1**<br><br>NUREG Bases markup p Insert page B 3.3-91, Insert Page B 3.3-94, and Insert Page B 3.3-98<br><br>Retyped Bases p B 3.3-86, B 3.3-89, and B 3.3-94 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Typographical errors (continued) | the correct JFD number has been provided (DB1 has been changed to DB2 for INSERT Function 1.d on the NUREG ITS markup for ITS 3.3.5.1); the word "occuring" has been changed to "occurring" in ITS 3.3.5.1 ASA, LCO, and Applicability Bases section for Function 2.h; Table "3.3.5.1" has been changed to "3.3.5.1-1", Table "3.3.5.2" has been changed to "3.3.5.2-1" and a semicolon has been added after the words "Function 1" in the ASA, LCO, and Applicability Bases section for ITS 3.3.5.1 Function 3.a; the word "contacts(s)" has been changed to "contact(s)" in SR 3.3.5.1.2 Bases; Table "3.3.5.1" has been changed to "3.3.5.1-1", Table "3.3.5.2" has been changed to "3.3.5.2-1" and a semicolon has been added after the words "Function 1" in the ASA, LCO, and Applicability Bases section for ITS 3.3.5.2 Function 1; the word "contacts(s)" has been changed to "contact(s)" in SR 3.3.5.2.2 Bases; a close parenthesis has been added to ITS 3.3.5.2 Reference 2; the words "And" and "For" have been decapitalized in ITS 3.3.5.2 Reference 3; the sign "<" has been changed to "≤" in ITS 3.3.6.1 DOC M14; a comma has been added to both Completion Times of ITS 3.3.6.1 Required Action A.1; the word "contacts(s)" has been changed to "contact(s)" in SR 3.3.6.1.2 Bases; "Note 7" has been changed to "Note 9" in ITS 3.3.6.2 DOC A7; the word "contacts(s)" has been changed to "contact(s)" in SR 3.3.6.2.2 Bases; the word "NOTES" has been changed to "NOTE" in ITS 3.3.7.1 Surveillance Requirements; four commas have been added to the ITS 3.3.7.3 Background section; two commas have been added to the ITS 3.3.7.3 LCO section; the word "Require" has been changed to "Required" and the word "the" has been deleted in the ITS 3.3.7.3 Actions;<br>CONTINUED ON NEXT PAGE | **Specification 3.3.5.1**<br><br>NUREG ITS markup p Insert Page 3.3-42<br><br>NUREG Bases markup p Insert page B 3.3-114, Insert Page B 3.3-115, and Insert Page B 3.3-136<br><br>Retyped Bases p B 3.3-113, B 3.3-114, and B 3.3-133<br><br>**Specification 3.3.5.2**<br><br>NUREG Bases markup p Insert Page B 3.3-141, Insert Page B 3.3-149, and B 3.3-151<br><br>Retyped Bases p B 3.3-139, B 3.3-146, and B 3.3-148<br><br>**Specification 3.3.6.1**<br><br>DOC M14 (DOCs p 10 of 25)<br><br>NUREG ITS markup p 3.3-52<br><br>NUREG Bases markup p Insert Page B 3.3-181<br><br>Retyped ITS p 3.3-47<br><br>Retyped ITS Bases p B 3.3-180<br><br>**Specification 3.3.6.2**<br><br>DOC A7 (DOCs p 2 of 12)<br><br>NUREG Bases markup p Insert Page B 3.3-194<br><br>Retyped ITS Bases p B 3.3-195<br><br>**Specification 3.3.7.1**<br><br>NUREG ITS markup p 3.3-73<br><br>Retyped ITS p 3.3-63<br><br>**Specification 3.3.7.3**<br><br>NUREG Bases markup p Insert Page B 3.3-219k, Insert Page B 3.3-219l, and Insert Page B 3.3-219n<br><br>Retyped ITS Bases p 3.3-210, B 3.3-211, and B 3.3-213 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Typographical errors (continued) | and the word "contacts(s)" has been changed to "contact(s)" in SR 3.3.8.2.1 Bases.) | **Specification 3.3.8.2**<br><br>NUREG Bases markup p Insert Page B 3.3-232<br><br>Retyped ITS Bases p B 3.3-228 |
| Consistency issues | Minor consistency issue corrections have been made. (The description of the SR 3.3.1.1.8 Channel Functional Test for Function 10 has been clarified in the Bases to be consistent with DOC LA8 statements (i.e., that it is performed by actually placing the reactor mode switch in the shutdown position) and the words "must be" are changed to "is" in the description of the ITS 3.3.1.1.8 Channel Functional Test for Function 2.b for consistency; the statement that physical inspection of the MSIV and main turbine stop valve position switches is performed in conjunction with the Channel Calibration has been deleted to be consistent with current plant practice and with the NUREG; the term "typical industry" has been deleted from the SR 3.3.1.1.15 Bases, consistent with other SR Bases; the words "or nonexistent" have been added to the ITS 3.3.1.2 Actions E.1 and E.2 Bases, consistent with the words in the D.1 and D.2 Bases; the words "In this event, the 12 hour Frequency is reasonable," has been changed to "Twelve hours is reasonable" in the SR 3.3.1.2.5/6 and SR 3.3.1.2.7 Bases to be consistent with similar statements in other ITS Section 3.3 SR Bases; the word "detectors" has been changed to "monitors" in SR 3.3.1.2.7 Bases to be consistent with the actual words in the SR; quotation marks have been placed around the title of LCO 3.1.1 in the ITS 3.3.2.1 Actions E.1 and E.2 Bases, consistent with their usage in the ITS Bases; the revision and date of ITS 3.3.2.1 Reference 3, which is also identified in the COLR, has been deleted and a statement that the Revision is specified in the COLR has been added, consistent with Specification 5.6.5; the word "May" has been changed to "October" to provide the correct date of ITS 3.3.2.1 Reference 9; the word "Note" has been changed to "NOTE" in ITS 3.3.2.2 Required Action C.1 and ITS 3.3.4.1 Required Action D.1 for consistency with the usage throughout the ITS; the term "typical industry" has been deleted and the word "cycles" changed to "cycle" in the SR 3.3.3.1.2 and SR 3.3.3.1.3 Bases, consistent with other SR Bases; the term "typical industry" has been deleted from the SR 3.3.3.2.3 Bases, consistent with other SR Bases;<br>CONTINUED ON NEXT PAGE | **Specification 3.3.1.1**<br><br>NUREG Bases markup p Insert Page B 3.3-29, Insert Page B 3.3-30, and B 3.3-32<br><br>Retyped ITS Bases p B 3.3-31, B 3.3-32, and B 3.3-35<br><br>**Specification 3.3.1.2**<br><br>NUREG Bases markup p B 3.3-39, B 3.3-42, and B 3.3-43<br><br>Retyped ITS Bases p B 3.3-42, B 3.3-46, and B 3.3-47<br><br>**Specification 3.3.2.1**<br><br>NUREG Bases markup p B 3.3-50, B 3.3-54, and B 3.3-55<br><br>Retyped ITS Bases p B 3.3-54, B 3.3-58, and B 3.3-59<br><br>**Specification 3.3.2.2**<br><br>NUREG ITS markup p 3.3-21<br><br>Retyped ITS p 3.3-21<br><br>**Specification 3.3.3.1**<br><br>NUREG Bases markup p B 3.3-73<br><br>Retyped ITS Bases p B 3.3-79<br><br>**Specification 3.3.3.2**<br><br>NUREG Bases markup p B 3.3-78<br><br>Retyped ITS Bases p B 3.3-85<br><br>**Specification 3.3.4.1**<br><br>NUREG ITS markup p 3.3-34<br><br>Retyped ITS p 3.3-30 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Consistency issues (continued) | the unit "feet" has been changed to "ft" for the Allowable Value of Table 3.3.5.1-1 Function 3.e for consistency with the writing of the unit throughout the ITS; the term "emergency diesel generator(s)" in Table 3.3.5.1-1 Note b has been changed to "emergency diesel generator subsystem" for consistency with the plant terminology in ITS 3.8.1 and 3.8.2; quotation marks have been placed around the LCO 3.5.2 title (and a long dash used) in the ITS 3.3.5.1 ASA, LCO, and Applicability Bases section, consistent with their use in the ITS Bases; the LCO 3.5.1, 3.5.2, and 3.8.1 titles have been deleted in the ITS 3.3.5.1 ASA, LCO, and Applicability section for Functions 1.a and 2.a, since they have previously been identified earlier in this section; the reference to LCO 3.5.2 for the HPCI System has been deleted from the ASA, LCO, and Applicability Bases section for ITS 3.3.5.1 Function 3.c, since LCO 3.5.2 does not provide any HPCI System requirements; the words "Function 2.e or 2.h, since this Function provides" in the ITS 3.3.5.1 Action B.1, B.2, and B.3 Bases have been changed to "Functions 2.e and 2.h, since these Functions provide" for consistency with the words in the ITS 3.3.5.1 Actions E.1 and E.2 Bases and grammatical correctness; "for" has been changed to "of" in the second Condition of ITS 3.3.6.1 Condition H for consistency with the wording of this type of Condition statement, and the two Conditions in Condition H have been swapped for consistency with their presentation in the ITS; the second Note of ITS 3.3.6.1 Surveillance Requirements has been rewritten to be in a similar format as a Note in ITS 3.3.5.1 Surveillance Requirements; periods have been deleted (the period at the end of the first Function) from the Functions 3.a and 4.a, Functions 3.b and 4.b, Function 3.c and 4.c, Functions 3.d, 3.e, 3.f, 3.g, 3.h, 3.i, 3.j, 4.d, 4.e, and 4.f, Functions 5.a, 5.b, and 5.c Headers in the ITS 3.3.6.1 ASA, LCO, and Applicability Bases section for consistency with other ITS Bases sections; a statement of which components are common to RPS has been added to ITS 3.3.6.1 Actions A.1 Bases, consistent with the LA DOCs; Function 1.f has been added to the the loss of function description for the MSL drain isolation in ITS 3.3.6.1 Action B.1 Bases, consistent with the description of the logic in the Background section; the term "typical industry" has been deleted from the SR 3.3.6.1.8 Bases, consistent with other SR Bases; a statement of which components are common to RPS has been added to ITS 3.3.6.2 Actions A.1 Bases, consistent with the LA DOCs; the words of SR 3.3.7.2.3 have been changed to be consistent with SR 3.3.6.1.6, an identical Surveillance Requirement;<br>CONTINUED ON NEXT PAGE | **Specification 3.3.5.1**<br><br>NUREG ITS markup p 3.3-42, 3.3-43, and 3.3-44<br><br>NUREG Bases markup p Insert Page B 3.3-107, B 3.3-109, B 3.3-116, and B 3.3-126<br><br>Bases JFD PA7 (Bases JFDs p 2 of 4)<br><br>Retyped ITS p 3.3-38, 3.3-39, and 3.3-40<br><br>Retyped ITS Bases p B 3.3-105, B 3.3-107, B 3.3-116, and B 3.3-124<br><br>**Specification 3.3.6.1**<br><br>NUREG ITS markup p 3.3-54 and 3.3-55<br><br>JFD PA5 (JFDs p 3 of 5)<br><br>NUREG ITS Bases markup p B 3.3-165, B 3.3-166, B 3.3-167, B 3.3-168, B 3.3-170, B 3.3-171, B 3.3-175, Insert Page B 3.3-176, B 3.3-180, and B 3.3-183<br><br>Retyped ITS p 3.3-49 and 3.3-50<br><br>Retyped ITS Bases p B 3.3-164, B 3.3-165, B 3.3-166, B 3.3-167, B 3.3-168, B 3.3-174, B 3.3-175, B 3.3-179, and B 3.3-183<br><br>**Specification 3.3.6.2**<br><br>NUREG Bases markup p B 3.3-191<br><br>Retyped ITS Bases p B 3.3-192 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Consistency issues (continued) | the words of SR 3.3.7.2.3 have been changed to be consistent with SR 3.3.6.1.6, an identical Surveillance Requirement; commas have been added to ITS 3.3.7.2 References 1 and 2; the words "Section" have been changed to "Chapter" in ITS 3.3.7.3 References 1 and 2; the term "emergency diesel generator(s)" in the LCO 3.3.8.1 Applicability has been changed to "emergency diesel generator subsystem" for consistency with the plant terminology in ITS 3.8.1 and 3.8.2; the LCO 3.8.1 and 3.8.2 titles have been deleted in the ITS 3.3.8.1 ASA, LCO, and Applicability section for Function 1, since they have previously been identified earlier in this section, and the word "LCO" has been added in front of 3.8.2 for consistency; and the words "pilot valve" have been added to the first paragraph of the ITS 3.3.8.2 Bases Background section, consistent with its usage throughout ITS 3.3.8.2 Bases.) | **Specification 3.3.7.2**<br><br>NUREG ITS markup p Insert Page 3.3-74b<br><br>NUREG Bases markup p Insert Page B 3.3-219k<br><br>Retyped ITS p 3.3-66<br><br>Retyped ITS Bases p B 3.3-209<br><br>**Specification 3.3.7.3**<br><br>NUREG Bases markup p Insert Page B 3.3-219p<br><br>Retyped ITS Bases p B 3.3-215<br><br>**Specification 3.3.8.1**<br><br>NUREG ITS markup p 3.3-75<br><br>NUREG Bases markup p B 3.3-222<br><br>Retyped ITS p 3.3-69<br><br>Retyped ITS Bases p B 3.3-219<br><br>**Specification 3.3.8.2**<br><br>NUREG Bases markup p B 3.3-227<br><br>Retyped ITS Bases p B 3.3-223 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Consistency issue | For consistency with ITS 3.1.8, Scram Discharge Volume Venbt and Drain Valves, ITS Table 3.3.1.1-1 Function 7 has been renamed the "Scram Discharge Volume" Water Level - High in lieu of "Scram Discharge Instrument Volume" Water Level - High. The detail that the Function monitors water level in the instrument portion of the SDV has been relocated to the Bases. | **Specification 3.3.1.1**<br><br>CTS markup p 8 of 16, 12 of 16, and 14 of 16<br><br>DOCs M2, M4, M6, and LA15 (DOCs p 7 of 25, 8 of 25, 9 of 25, and 16 of 25)<br><br>NUREG markup p 3.3-9<br><br>JFD DB7 (JFDs p 3 of 5)<br><br>NUREG Bases markup p B 3.3-1, B 3.3-16, and B 3.3-17<br><br>Bases JFD DB9 (Bases JFDs p 3 of 4)<br><br>Retyped ITS p 3.3-8<br><br>Retyped ITS Bases p B 3.3-3 and B 3.3-18 |
| Consistency issue | The Bases concerning the Response Time Tests have been modified to reflect current plant licensing basis (i.e., the generic words in the Bases have been replaced with plant specific words). | **Specification 3.3.1.1**<br><br>NUREG Bases markup p B 3.3-32<br><br>Retyped ITS Bases p B 3.3-35<br><br>**Specification 3.3.6.1**<br><br>NUREG Bases markup p B 3.3-183<br><br>Retyped ITS Bases p B 3.3-183 |
| Consistency issue | ITS SR 3.3.2.1.5, the 92 day RBM Channel Calibration Surveillance, is modified by the addition of a Note that excludes the recirculation loop flow signal portion of the channel. CTS Table 4.1-2, "Flow Bias Signal," requires an "internal power and flow test with standard pressure source" calibration on a refueling interval. This is covered by ITS SR 3.3.1.1.12 (RPS). This flow biased signal provides input to both the APRM Neutron Flux – High (Flow Biased) RPS scram Function and to the RBM – Upscale control rod block Function. CTS 3.2.C does not have a specific flow bias signal line item, thus, the current plant practice is that the calibration required by CTS Table 4.1-2 covers the RBM requirements, as well as the RPS requirements, of the flow bias signal (i.e., there is no flow signal calibration in the RBM channel cal procedure). Therefore, the RBM Channel Calibration requirement in SR 3.3.2.1.5 is modified to exclude the flow bias signal. The ITS Bases clearly identifies that SR 3.3.1.1.12 covers the flow bias signal. | **Specification 3.3.2.1**<br><br>NUREG ITS markup p 3.3-19<br><br>JFD CLB1 (JFDs p 1 of 3)<br><br>NUREG Bases markup p B 3.3-54<br><br>Retyped ITS p 3.3-18<br><br>Retyped ITS Bases p B 3.3-57 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Consistency issue | The NUREG for Specification 3.3.7.1 formats the Bases such that the ASA, LCO, and Applicability sections are combined. This is similar to other like Instrumentation Bases, where there are multiple Functions covered by the Specification. However, for the FitzPatrick ITS 3.3.7.1, only one Function is covered by the LCO statement. Therefore, for consistency with NUREG Instrumentation LCOs where only one Function is covered, the individual ASA, LCO, and Applicability sections are separated. | Specification 3.3.7.1<br><br>NUREG Bases markup p B 3.3-208, B 3.3-211, and B 3.3-212<br><br>Retyped ITS Bases P B 3.3-198, B 3.3-199, and B 3.3-200 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Editorial | The description of the reference point for the low water level Allowable Value has been modified for clarity. | **Specification 3.3.1.1**<br><br>NUREG Bases markup p Insert Page B 3.3-14<br><br>Retyped ITS Bases p B 3.3-15<br><br>**Specification 3.3.4.1**<br><br>NUREG Bases markup p Insert Page B 3.3-94<br><br>Retyped ITS Bases p B 3.3-89<br><br>**Specification 3.3.5.1**<br><br>NUREG Bases markup p Insert Page B 3.3-109, Insert Page B 3.3-113, Insert Page B 3.3-115, Insert Page B 3.3-116, Insert Page B 3.3-120, and Insert Page B 3.3-122<br><br>Retyped ITS Bases p B 3.3-106, B 3.3-112, B 3.3-114, B 3.3-116, B 3.3-119, and B 3.3-121<br><br>**Specification 3.3.5.2**<br><br>NUREG Bases markup p Insert Page B 3.3-141 and Insert Page B 3.3-142<br><br>Retyped ITS Bases p B 3.3-139 and B 3.3-140<br><br>**Specification 3.3.6.1**<br><br>NUREG Bases markup p Insert Page B 3.3-158, Insert Page B 3.3-164a, Insert Page B 3.3-172, and Insert Page B 3.3-174a<br><br>Retyped ITS Bases p B 3.3-156, B 3.3-163, B 3.3-169, B 3.3-171, and B 3.3-172<br><br>**Specification 3.3.6.2**<br><br>NUREG Bases markup p Insert Page B 3.3-188<br><br>Retyped ITS Bases p B 3.3-189 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Editorial | The reference to a loss of main condenser vacuum event in the ASA, LCO, and Applicability section of ITS 3.3.1.1 Function 8 has been deleted, since the loss of main condenser vacuum event is a subset of (i.e., it is already covered by) the turbine trip event, which is being maintained in this section of the Bases. | **Specification 3.3.1.1**<br><br>NUREG Bases markup p B 3.3-17<br><br>Retyped ITS Bases p B 3.3-19 |
| Editorial | The proper References have been provided for the Bases of ITS 3.3.2.1, ITS 3.3.3.2, ITS 3.3.5.1, ITS 3.3.7.2, and ITS 3.3.8.1. | **Specification 3.3.2.1**<br><br>NUREG Bases markup p B 3.3-46 and B 3.3-54<br><br>Bases JFD DB9 (Bases JFDs p 2 of 3)<br><br>Retyped ITS Bases p B 3.3-50 and B 3.3-58<br><br>**Specification 3.3.3.2**<br><br>NUREG Bases markup p B 3.3-75, B 3.3-77, and B 3.3-79<br><br>Retyped ITS Bases p B 3.3-81, B 3.3-83, and B 3.3-85<br><br>**Specification 3.3.5.1**<br><br>NUREG Bases markup p B 3.3-138<br><br>Retyped ITS Bases p B 3.3-135<br><br>**Specification 3.3.7.2**<br><br>NUREG Bases markup p Insert Page B 3.3-219k<br><br>Retyped ITS Bases p B 3.3-209<br><br>**Specification 3.3.8.1**<br><br>NUREG Bases markup p B 3.3-220 and Insert Page B 3.3-220<br><br>Retyped ITS Bases p B 3.3-216 |
| Editorial | The proper reason for the Allowable Value of ITS 3.3.4.2.a has been provided. Also, the word "channel(s)" has been changed to "channels", consistent with the JAFNPP design. | **Specification 3.3.4.1**<br><br>NUREG Bases markup p B 3.3-94, Insert Page B 3.3-94, and B 3.3-99<br><br>Retyped ITS Bases p B 3.3-89 and B 3.3-95 |

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Editorial | The decription in the ITS 3.3.5.1 Bases Background section of how the ADS Initiation Timers are reset has been clarified (consistent with plant design). | **Specification 3.3.5.1**<br><br>NUREG Bases markup p B 3.3-105<br><br>Retyped ITS Bases B 3.3-103 |
| Editorial | The Allowable Value shown in ITS 3.3.6.1 Discussion of Change L1 has been deleted, since it was incorrect and not needed to properly describe the change. (Note: The Allowable Value in the ITS is correct). | **Specification 3.3.6.1**<br><br>DOC L1 (DOCs p 17 of 25) |
| Editorial | A more accurate UFSAR reference has been provided (the actual Table number has been identified). | **Specification 3.3.6.1**<br><br>NUREG Bases markup p B 3.3-184<br><br>Retyped ITS Bases p B 3.3-183 |
| Editorial | The description in the ITS 3.3.6.2 Bases Background section of what components the secondary containment isolation instrumentation initiates has been clarified (consistent with plant design). Also, the statement that "Certain instrumentation Functions are retained...discussion" in the ITS 3.3.6.2 Bases Applicable Safety Analyses section has been deleted since all the Functions are retained due to Criterion 3. | **Specification 3.3.6.2**<br><br>NUREG Bases markup p Insert Page B 3.3-185 and B 3.3-186<br><br>Retyped ITS Bases p B 3.3-186 and B 3.3-187 |
| Editorial | The description in the ITS 3.3.7.1 ACTIONS A.1 and A.2 Bases that the method used to place the CREVAS System in operation must ensure that it restarts automatically upon restoration of power following a loss-of-power event has been deleted, since the CREVAS System is a manually initiated system (as described in the Background section of the Bases. | **Specification 3.3.7.1**<br><br>NUREG Bases markup p B 3.3-215<br><br>Retyped ITS Bases p B 3.3-200 |
| Editorial | The first sentence of the ITS 3.3.7.3 LCO section has been divided into two sentences with minor editorial changes for clarity. | **Specification 3.3.7.3**<br><br>NUREG Bases markup p Insert Page B 3.3-219l<br><br>Retyped ITS Bases p B 3.3-211 |
| Technical change | The Frequency for performing the SRM/IRM overlap Surveillance has been changed from "Prior to withdrawing SRMs from the fully inserted position" to "Prior to fully withdrawing SRMs", since it is not always possible to obtain proper overlap prior to reaching the SRM rod block setpoint with the SRMs fully inserted. This change is consistent with the most recent BWR ITS submittals (LaSalle and NMP2). | **Specification 3.3.1.1**<br><br>DOC M7 (DOCs p 9 of 25)<br><br>NUREG markup p 3.3-4<br><br>JFD DB10 (JFDs p 4 of 5)<br><br>NUREG Bases markup p B 3.3-28<br><br>Bases JFD DB10 (Bases JFDs p 3 of 4)<br><br>Retyped ITS p 3.3-4 and B 3.3-30 |

SUMMARY OF CHANGES TO ITS SECTION 3.3 - REVISION J

| Source of Change | Summary of Change | Affected Pages |
|---|---|---|
| Technical change | The SRM signal to noise ratio in SR 3.3.2.1.4 has been changed from "3:1" to "2:1" to be consistent with the design documents. | **Specification 3.3.1.2**<br><br>DOC A2 (DOCs p 1 of 7)<br><br>NUREG ITS markup p 3.3-13<br><br>JFD CLB1 (JFDs p 1 of 2)<br><br>Retyped ITS p 3.3-13 |
| Technical change | The oxygen analyzer range has been changed from "0% to 25%" to "0% to 30%", based on a recent UFSAR update. | **Specification 3.3.3.1**<br><br>NUREG Bases markup p B Insert Page 3.3-68<br><br>Retyped ITS Bases p B 3.3-74 |
| Technical change | For Core Spray, LPCI, and HPCI Pump discharge flow low Functions in ITS 3.3.5.1 (Functions 1.e, 2.g, and 3.f), the Channel Calibration (including the Channel Functional Test) Frequency is being changed from the current 92 day Frequency to a 24 month Frequency. This 24 month Frequency is consistent with (or more frequent than) current practice (The Channel Calibration requirement for these Functions is not in the CTS). A review of maintenance history has been performed and shows that a 24 month Frequency for these Functions is acceptable. In addition, the drift analyses in the setpoint calculations support the 24 month Frequency. | **Specification 3.3.5.1**<br><br>CTS markup p 10 of 15<br><br>DOC M2 (DOCs p 8 of 13)<br><br>NUREG ITS markup p 3.3-42, 3.3-44, and 3.3-45<br><br>JFD DB7 (JFDs p 3 of 4)<br><br>Retyped ITS p 3.3-38, 3.3-40, and 3.3-41 |
| NRC question | DOC L1 has been changed as requested by the NRC reviewer. Information has been provided that with the channel in trip, the ESW System will still perform satisfactorily. | **Specification 3.3.7.3**<br><br>DOC L1 (DOCs p 3 of 5) |

Specification 3.3.1.1

3.3.1.1-1

TABLE 4.1-1 (cont'd)

## REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

| Minimum No. of Operable Instrument Channels Per Trip System (Notes 1 and 2) | Trip Function | Trip Level Setting | Mode in Which Function Must Be Operable: Refuel (Note 7) | Mode in Which Function Must Be Operable: Startup | Mode in Which Function Must Be Operable: Run | Total Number of Instrument Channels Provided by Design for Both Trip Systems | Action (Note 3) |
|---|---|---|---|---|---|---|---|
| 2 | Reactor High Pressure | ≤ 1060 psig | X | X | X | 4 | |
| 2 | Drywell High Pressure (Note 16) | ≤ 2.7 psig | X | X | X | 4 | |
| 2 | Reactor Low Water Level (Note 18) | ≥ 177 in. above TAF | X | X | X | 4 | |
| 2 2 | High Water Level in Scram Discharge Volume | ≤ 34.5 gallons per instrument volume (Note 4) | X | X | X | 8 | |
| | Main Steam Line Isolation Valve Closure | ≤ 15% valve closure | | | X (Note 6) | 8 | |
| 2 | Turbine Control Valve Fast Closure | 500 ≤ P ≤ 850 psig; Control oil pressure between fast closure solenoid and disc dump valve | | | X (Note 5) | 4 | |
| 4 | Turbine Stop Valve Closure | ≤ 10% valve closure | | | X (Notes 5 & 6) | 8 | |

Amendment No. ~~19, 30, 43, 72, 87, 88, 134, 162, 227,~~ 265



Page 8 of 16

REVISION J

Specification 3.3.1.1

A1

JAFNPP

TABLE 4.1-1 3.3.1.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION TEST REQUIREMENTS

| | Trip Function | Group (Note 2) L6 | Functional Test | Functional Test Frequency (Note 3) A10 | Instrument Check channel |
|---|---|---|---|---|---|
| [10] | Mode Switch in Shutdown | A | Place Mode Switch in Shutdown LA8 | H SR 3.3.1.1.11 | NA add proposed SR 3.3.1.1.13 M4 |
| [11] | Manual Scram A17 | A | Trip Channel and Alarm L7 | Q SR 3.3.1.1.8 | NA |
| [SR3.3.1.1.4] | RPS Channel Test Switch M7 add proposed SR 3.3.1.1.5 | A | Trip Channel and Alarm (Note 4) | W (Note 1) LA9 | NA |
| [1.a] | IRM High Flux | C | Trip Channel and Alarm (Note 4) | S/U and W (Note 5) | NA 12 Hours M5 add proposed SR 3.3.1.1.1 |
| [1.b] | IRM Inoperative M4 add proposed SR 3.3.1.1.13 | C | Trip Channel and Alarm (Note 4) | S/U and W (Note 5) L10 SR 3.0.4 SR 3.3.1.1.3 | NA add proposed note |
| | APRM | | | | |
| [2.c] | High Flux | B | Trip Output Relays (Note 4) | Q SR 3.3.1.1.8 | NA |
| [2.d] | Inoperative | B | Trip Output Relays (Note 4) | Q SR 3.3.1.1.3 | NA |
| [2.b] | Flow Biased High Flux | B | Trip Output Relays (Note 4) | Q | NA M5 SR 3.3.1.1.6 |
| [2.a] | High Flux in Startup or Refuel | C | Trip Output Relays (Note 4) | S/U and W (Note 5) SR 3.0.4 | NA |
| [3] | Reactor High Pressure | B | Trip Channel and Alarm (Note 4) | Q | D 12 hours M6 |
| [6] | Drywell High Pressure | B | Trip Channel and Alarm (Note 4) | Q | D SR 3.3.1.1.1 |
| [4] | Reactor Low Level | B | Trip Channel and Alarm (Note 4) | Q SR 3.3.1.1.8 | D |
| [7b] | High Water Level in Scram Discharge Instrument Volume | A | Trip Channel | Q (Note 6) LA11 | NA |
| [7a] | High Water Level in Scram Discharge Instrument Volume LA15 | B | Trip Channel and Alarm (Note 4) | D A15 | D |

J

Specification 3.3.1.1 (A1)



3.3.1.1-1

TABLE 4.1-2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

add proposed SR 3.3.1.1.9 (A20)

Function

(A10) (M14)

| | Instrument Channel | Group (1) | Calibration | Frequency (2) |
|---|---|---|---|---|
| [1.a] | IRM High Flux | C | Comparison to APRM on Controlled Shutdowns [SR 3.3.1.1.2] (M10) (L12) add proposed Note | W [SR 3.3.1.1.6] add proposed Note (M14) |
| | APRM High Flux Output Signal (add proposed SR 3.3.1.1.9 for Function 2.a, 2.b, 2.c) (A20) | B | Heat Balance | ~~D~~ 7 days (L13) |
| [2.b] | Flow Bias Signal (SR 3.3.1.1.7) | B | Internal Power and Flow Test with Standard Pressure Source | R [SR 3.3.1.1.12, including Note 3 to SR 3.3.1.1.9] |
| [2.] | LPRM Signal | B | [2.b, 2.c] (L7) | Every 1000 MWD/T average core exposure |
| [3] | High Reactor Pressure | B | Standard Pressure Source | (Note 6) [SR 3.3.1.1.10, SR 3.3.1.1.12] |
| [6] | High Drywell Pressure | B | Standard Pressure Source | (Note 6) [SR 3.3.1.1.10, SR 3.3.1.1.12] |
| [4] | Reactor Low Water Level | B | Standard Pressure Source | (Note 6) [SR 3.3.1.1.10, SR 3.3.1.1.12] |
| [7.b] | High Water Level in Scram Discharge ~~Instrument~~ Volume | A | Water Column ~~(Note 5)~~ (LA11) | R ~~(Note 5)~~ [SR 3.3.1.1.12] |
| [7.a] | High Water Level in Scram Discharge ~~Instrument~~ Volume (LA15) | B | Standard Pressure Source | Q [SR 3.3.1.1.9] |
| [5] | Main Steam Line Isolation Valve Closure | A | ~~(Note 4)~~ (M12) | ~~(Note 4)~~ [SR 3.3.1.1.12] |
| [8][9] | Turbine First Stage Pressure Permissive (LA12) | B | Standard Pressure Source (L7) | (Note 6) [SR 3.3.1.1.10, SR 3.3.1.1.12] |

(L6)

add SR 3.3.1.1.6 to Function 2.a (M14)

(J)

Amendment No. ~~42, 43, 62, 75, 89, 136, 183, 207, 233~~, 257





Revision ~~B~~ J.

ADMINISTRATIVE CHANGES

A19 (continued)

"Allowable Values". Since the instrumentation will be declared inoperable at the same numerical value, this change is considered administrative. Any changes to any "Trip Level Setting" or "limiting safety system trip settings" in the CTS will be discussed below. This change is consistent with NUREG-1433, Revision 1.

A20 The CTS does not have a specific CHANNEL CALIBRATION requirement for the APRM and IRM RPS Functions. However, the CTS does have a 92 day CHANNEL CALIBRATION requirement for the APRM and IRM Control Rod Block Functions. Therefore, consistent with this CTS requirement and with current practice, a Surveillance Requirement is included as ITS SR 3.3.1.1.9 to perform a CHANNEL CALIBRATION on IRM Function 1.a and APRM Functions 2.a, 2.b, and 2.c every 92 days.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS Table 3.1-1, Note 4, that allows the Scram Discharge Volume High Function to be bypassed when the mode switch is in refuel or shutdown, is being deleted. ITS Table 3.3.1.1-1 Function 7 footnote (a) requires this Function to be OPERABLE in MODE 5 whenever any control rod is withdrawn from a core cell containing one or more fuel assembles. This will ensure that if a scram occurs the control rod insertion will not be hindered by the water level in the scram discharge volume being too high. When the reactor mode switch is in shutdown, the control rods can not be withdrawn, therefore this scram function is not required. This change is consistent with the requirements of NUREG-1433, Revision 1. This change constitutes a more restrictive requirement, and is not considered to result in any reduction to safety.

M2 CTS Table 3.1-1 requires 3 channels of Scram Discharge Volume High Water Level to be OPERABLE in each Trip System. In the ITS, the Scram Discharge Water level Functions have been divided into Table 3.3.1.1-1 Functions 7.a and 7.b. Both Function 7.a (Scram Discharge Volume Water Level - Differential Pressure Transmitter/Trip Unit) and Function 7.b (Level Switch) require 2 channels to be OPERABLE in each Trip System. This change is more restrictive since the required number of channels has been increased from 3 channels to 4 channels in each Trip System. This change is consistent with NUREG-1433, Revision 1. J

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

M3 CTS Table 3.1-1 requires 4 channels of Main Steam Line Isolation Valve Closure to be OPERABLE in each Trip System. In the ITS, Table 3.3.1.1-1 Functions 5 (Main Steam Isolation Valve – Closure) require 8 channels to be OPERABLE in each Trip System. This change is more restrictive since the required number of channels has been increased from 4 channels to 8 channels in each Trip System. This change is consistent with NUREG-1433, Revision 1.

M4 ITS SR 3.3.1.1.13 adds the requirement to perform Logic System Functional Tests every 24 months for the following Functions:

IRM Neutron Flux – High (MODE 2 and MODE 5(a))
IRM Inop (MODE 2 and MODE 5(a))
APRM Neutron Flux – High (Startup) (MODE 2)
APRM Neutron Flux – High (Flow Biased)
APRM Neutron Flux – High (Fixed)
APRM Inop (MODE 1 and MODE 2)
Reactor Pressure – High
Reactor Vessel Water Level – Low (Level 3)
Main Steam Isolation Valve – Closure
Drywell Pressure – High
SDV Water Level – High (MODE 1, MODE 2, and MODE 5(a)) 13
Turbine Stop Valve – Closure
Turbine Control Valve Fast Closure, EHC Trip Oil Pressure-Low
Reactor Mode Switch – Shutdown Position (MODE 1, MODE 2, and MODE 5(a))
Manual Scram (MODE 1, MODE 2, and MODE 5(a))

The addition of new requirements (Surveillances) to the current Technical Specifications constitutes a more restrictive change. The added testing is currently being performed at JAFNPP in accordance with the guidelines of GL-96-01 (Testing of Safety-Related Logic) therefore this change will not add any additional testing. This change is consistent with NUREG-1433, Revision 1. This change is not considered to result in any reduction to safety.

M5 ITS SR 3.3.1.1.1, adds the requirement to perform Channel Checks every 12 hours for the Functions listed below:

IRM Neutron Flux – High (MODE 2 and MODE 5(a))
APRM Neutron Flux – High (Startup) (MODE 2)
APRM Neutron Flux – High (Fixed) (MODE 1)
APRM Neutron Flux – High (Flow Biased) (MODE 1)

TECHNICAL CHANGES - MORE RESTRICTIVE

M5 (continued)

The addition of new requirements (Surveillances) to the current Technical Specifications constitutes a more restrictive change necessary to ensure the RPS Functions are maintained Operable. This change is consistent with NUREG-1433, Revision 1. This change is not considered to result in any reduction to safety.

M6 ITS SR 3.3.1.1.1, increases the frequency for performing the Channel Checks in CTS Table 4.1-1 from the current Daily to every 12 hours for the Functions listed below:

Reactor Pressure – High
Drywell Pressure – High
Reactor Vessel Water Level – Low (Level 3)
Scram Discharge Volume Water Level – High
(DP transmitter/trip unit)
Turbine First Stage Pressure Permissive (see LA12)

This change to the requirements (Surveillances) of the current Technical Specifications constitutes a more restrictive change necessary to ensure the RPS Functions are maintained Operable. This change is consistent with NUREG-1433, Revision 1. This change is not considered to result in any reduction to safety.

M7 ITS SR 3.3.1.1.5 was added to verify SRM and IRM channels overlap prior to fully withdrawing SRMs. This change to the requirements (Surveillances) of the current Technical Specifications constitutes a more restrictive change necessary to ensure the RPS Functions are maintained Operable.

M8 CTS 4.1.A specifies that the response time of the reactor protection system trip functions listed shall be demonstrated to be within its limit once per 24 months. Each test shall include at least one channel in each trip system. All channels in both trip systems shall be tested within two test intervals. In ITS SR 3.3.1.1.15 the RPS RESPONSE TIME test must be performed every 24 months on a STAGGERED TEST BASIS. Note 3 of this SR specifies that "n" equals 2 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. Therefore, SR 3.3.1.1.15 will require all channels requiring response time testing to be tested in two (2) surveillance intervals. This change is more restrictive since at least eight (8) ITS 3.3.1.1 Function 5 (Main Steam Isolation Valve – Closure) channels and four (4) ITS 3.3.1.1 Function 8 (Turbine Stop Valve – Closure) channels must be tested each interval

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA13 (continued)

ensure the RPS instrumentation is OPERABLE. The requirements of ITS 3.3.1.1 which require the RPS instrumentation to be OPERABLE and the definition of OPERABILITY suffice. As such, these details are not required to be in the ITS to provide adequate protection of public health and safety. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the ITS.

LA14 The details described in CTS 4.1.A footnote * that state that the sensor is eliminated from response time testing for the RPS actuation logic circuits for Reactor High Pressure and Reactor Water Level-Low CTS functions is relocated to the Bases. These operational details are not necessary to ensure the RPS instrumentation is OPERABLE. The requirements of ITS 3.3.1.1 which require the RPS instrumentation to be OPERABLE and the definition of OPERABILITY suffice. As such, these details are not required to be in the ITS to provide adequate protection of public health and safety. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the ITS. In addition, the relocation of these details to the Bases is consistent with TSTF 332, R1.

LA15 The details in CTS Tables 3.1-1, 4.1-1, and 4.1-2 that the measured level for the Scram Discharge Volume (SDV) High Level scram Function is in the instrument volume portion of the SDV is proposed to be relocated to the Bases. These details of system design are not necessary to ensure the SDV Water Level – High Function is Operable. The requirement in ITS LCO 3.3.1.1 that the RPS instrumentation be Operable and the Allowable Value specified in ITS Table 3.3.1.1-1, Functions 7.a and 7.b, are adequate to ensure the Allowable Value is properly maintained. 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 Bases Control Program described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS Table 3.1-1 Note 7 Applicability (reactor is subcritical, fuel is in the vessel and the reactor temperature is less than 212°F) for the Mode Switch in Shutdown, Manual Scram, and IRM High Flux, is being relaxed. ITS Table 3.3.1.1-1, footnote (a), establishes requirements for when in MODE 5 (Refuel) with any control rod withdrawn from a core cell containing one or more fuel assemblies. This change also proposes to relax the Applicability for the IRM Inoperative Function in CTS

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 (continued)

Table 3.1-1 from when the mode switch is in Refuel to MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. These changes in the Applicability are consistent with the Applicability requirements for the scram discharge volume high level Functions as indicated in Note 7. This change does not impact the safety of the plant or any of the safety analysis assumptions. The design function, of the RPS Functions, is to shutdown the reactor when required by initiating a reactor scram. This is only necessary when control rods are withdrawn. Control rods withdrawn from a core cell containing no fuel assemblies do not affect the reactivity of the core. With all the rods inserted, the Shutdown Margin Requirements (LCO 3.1.1) and the required one-rod-out interlock (LCO 3.9.2) ensure that no scram is necessary. The Actions for inoperable equipment in MODE 5 are also revised to be consistent with the proposed Applicability. Since all control rods are required to be fully inserted during fuel movement (LCO 3.9.3), the proposed applicable conditions cannot be entered while moving fuel. The only possible core alteration is control rod withdrawal which is adequately addressed by the proposed actions. This change is consistent with NUREG-1433, Revision 1. Special Operations ITS 3.10.4 will allow a single control rod to be withdrawn in MODE 4 by allowing the Reactor Mode Switch to be in the Refuel position. Therefore, the IRM MODE 4 RPS requirements have been included in ITS 3.10.4.

L2 CTS Table 3.1-1 Note 3.A action time, to reach MODE 3 (all rods inserted) in 4 hours, is proposed to be extended. Proposed ITS 3.3.1.1 ACTION G requires being in MODE 3 within 12 hours. This provides the necessary time to shutdown in a controlled and orderly manner that is within the capabilities of the plant, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a plant upset that could challenge safety systems. This time is consistent with NUREG-1433, Revision 1.

L3 CTS Table 3.1-1 Note 3.A (for Mode Switch in Shutdown, Manual Scram, IRM High Flux, IRM Inoperative, and High Water Level in Scram Discharge Volume Functions) requires the insertion of all operable control rods within 4 hours if the requirements of Table 3.1-1 are not met. ITS 3.3.1.1 ACTION H will require, in MODE 5 for the above listed Functions, control rods in core cells containing one or more fuel assemblies to be inserted if ACTION A, B, or C cannot be performed within the required Completion Times. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core cells and are, therefore, not required to be inserted. The removal of the four fuel bundles surrounding a control rod very significantly reduces the

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 (continued)

reactivity worth of the associated control rod to the point where removal of that rod no longer has the potential to cause a reactivity excursion. This is reflected in the proposed definition of Core Alterations. This change is consistent with NUREG-1433, Revision 1.

L4 CTS Table 3.1-1 requirements,for APRM Neutron Flux-Startup (Note 7), APRM Inoperative during MODE 5 operations, and CTS 2.1.A.1.b requirements for APRM Neutron Flux scram during refuel are proposed to be deleted. Amendments 41 and 7 to Limerick Generating Station Units 1 and 2 (NPF-39 and NPF-85), respectively, issued July 30, 1990, eliminated APRM RPS trip OPERABILITY requirements during MODE 5, other than during SDM demonstrations. This remaining requirement is therefore moved into the SHUTDOWN MARGIN demonstration Special Operation Technical Specification (ITS 3.10.8).

A JAF plant specific analysis which justifies the proposed CTS changes described above is provided below. The JAF analysis presented below is consistent with the evaluation presented in the License Amendements for the Limerick Units.

The proposed CTS changes remove the requirements for APRM operability while the plant is in the Refuel Mode. To assess the impact of the proposed change on safety and the design bases accidents, an examination of those systems and mechanisms which contribute to safe operation while the plant is in the Refuel Mode is presented below. Each of these systems and mechanisms contribute to the defense-in-depth design and operation. This examination demonstrates that the current APRM operability requirement is unnecessary to maintain this defense-in-depth.

The SRM and IRM are subsystems of the Neutron Monitoring System (NMS). The purpose of these subsystems is to monitor neutron flux levels and provide, as appropriate, trip signals to the Reactor Protection System (RPS).

The SRM subsystem is composed of four detectors that are inserted into the core during shutdown and refuel conditions. Although the SRM subsystem is not safety-related, it is important to plant safety. During refueling operations, the plant operators use the SRMs to ensure that neutron flux remains within an acceptable range. Also, plant operators can monitor the SRMs for increases in neutron flux which may indicate that the reactor is approaching criticality. The SRMs are required by TS to be operational in the Refuel Mode (CTS 3.3.B.4, 4.3.B.4, 4.10.B and 3.10.B.2) (ITS Table 3.3.1.2-1).

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 (continued)

The IRM subsystem is composed of eight detectors that are inserted into the core. The IRM is a safety related subsystem. The IRM is a five-decade instrument with ten ranges that are ranged up during normal power increases. The IRMs are designed to monitor neutron flux levels at a local core location and provide protection against local criticality events caused by control rod withdrawal and fuel insertion errors. The IRMs monitor neutron flux levels from the upper portion of the SRM range to the lower portion of the APRM range. In terms of rated reactor power, the IRMs range from about 10E-4% of rated reactor power to greater than 15% of rated reactor power. The IRMs provide a scram function at ≤ 120 of a 125 division scale. The safety design bases of the IRM subsystem is to generate trip signals to prevent fuel damage resulting from anticipated or abnormal operational transients that could possibly occur while operating in the intermediate power range. The IRMs are required by TS to be operational in the Refuel Mode (CTS 2.1.A.1.a; Table 3.3-1, Item 3; Table 4.1-1, Item 4 and Table 4.1-2, Item 1)(ITS Table 3.3.1.1-1, Function 1a)

There are various levels of control to prevent inadvertent reactor criticality and fuel damage during refueling operations. These levels of control include the following:

1. Licensed plant operators are trained to operate equipment and follow approved procedures.

2. Plant approved refueling and maintenance procedures specify core alteration steps.

3. SRMs indicate the potential for reactor criticality by monitoring neutron flux levels.

4. Refueling interlocks prevent the withdrawal of more than one control rod and prevent the insertion of fuel assemblies into the core unless all control rods are fully inserted (except as permitted by CTS Section 3.10, "Core Alterations" and ITS 3.10.6, "Multiple Control Rod Withdrawal - Refueling").

5. The IRMs provide an indication of local power. IRMs provide a scram signals on high neutron flux levels.

The APRMS are not necessary for safe operation of the plant during refueling because the IRMs will generate an RPS scram if neutron flux increases to the applicable setpoint. The IRMs are required by TS to be operational in the Refuel Mode. The IRMs are a safety-related subsystem

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 (continued)

of the NMS and are designed to indicate and respond to neutron flux increases at local core locations. The APRMs are designed to monitor and respond to a core average neutron flux level. The most likely reactivity insertion transient expected during refueling would be a core alteration type event, e.g., control rod withdrawal or fuel assembly insertion into the core. A core alteration event would result in a local core criticality transient readily detected by the IRMs and/or SRMs.

The IRM subsystem is designed and calibrated to respond to a neutron flux level that is significantly less than the flux level monitored by the APRMs. For example, during refueling, when the IRMs are on their most sensitive range, the IRMs will generate a scram signal at less than 0.01% core average power while the APRMs will generate a scram signal at ≤15% core average power. The IRM subsystem acts as a backup protection system to the Refueling Interlocks (RIs) during refueling.

RIs are required to be operational during refueling operations (CTS 3.10.A.1) (ITS 3.9.1 & 3.9.2). The purpose of the RIs is to restrict the movement of the control rods and the operation of the refueling equipment to reinforce operational procedures that prevent the reactor from becoming critical during refueling operations. RIs will prevent the withdrawal of a control rod if the refueling platform is over the core. Also, the RIs require an "all-rods-in" signal before allowing the refueling platform to go over the core.

TS and plant operating procedures allow only one control rod to be withdrawn or removed at a time while the mode switch is in "Refuel" (except as permitted by CTS section 3.10, "Core Alterations" and ITS 3.10.6, "Multiple Control Rod Withdrawal - Refueling"). The core loading pattern is designed to ensure that the core is subcritical by a specified margin with the most reactive control rod at the full out position. Withdrawal of one control rod would not cause criticality and the event would not result in an APRM response.

The design of the control rod drive system reduces the probability of a control rod error during refueling. For example, the latching action of the collet finger assembly serves to lock the index tube in place. The velocity limiter physically prevents the control blade from being removed from the core with fuel in place.

The James A. FitzPatrick Final Safety Analysis Report (FSAR) Section 14.5.4, "Events Resulting in a Positive Reactivity Insertion," evaluated the potential for a control rod withdrawal error and fuel assembly

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 (continued)

insertion error during refueling. The FSAR concludes that the above scenarios are adequately precluded by refueling interlocks, core design, and control rod hardware design. However, should operator errors, followed by equipment malfunctions, result in an inadvertent criticality event, necessary safety actions (a scram) will be taken prior to violation of a safety limit. Specifcally, the IRMs would provide a scram function as appropriate.

The hypothetical question arises as to whether the APRM subsystem (if operable) would indicate and scram the control rods on a high neutron flux level before the operable IRMs would respond to the event. The answer is that a neutron flux transient would be observed by the IRMs before the APRM electronics would detect the event. The core coupling is such that a local criticality event would immediately be transmitted throughout the core and would be detected by the operable IRMs. The IRMs would be on scale before the APRMs detected the event because the IRMs are designed and calibrated to be more sensitive to neutron flux than the APRMs.

In summary, the APRMs are not necessary for safe operation of the plant while in the Refuel Mode for the following reasons:

1. The IRMs are a safety-related subsystem of the NMS and are required by TS to be operable in the Refuel Mode. The IRMs will generate an RPS Scram if the neutron flux increases to the applicable setpoint.

2. The IRMs and SRMs are designed and calibrated to be more sensitive to neutron flux than the APRMs.

3. The IRMs are designed to monitor local core events while the APRMs provide a measure of core average power condition. The IRMs can monitor and react to the reactivity events expected during refueling, i.e., control rod withdrawal or fuel insertion.

4. The IRMs would detect and respond (reactor scram) to an inadvertent criticality event before the APRMs would provide a trip function.

5. The withdrawal of only one control rod in the Refuel Mode is permitted by the "one-rod-out" interlock while in "Refuel". The core is designed to be subcritical with one rod out.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 (continued)

6. The withdrawal of a second control rod or inadvertent insertion of a fuel bundle in the Refuel Mode is precluded by refueling interlocks, refueling procedures, and administrative controls.

7. The APRMs are required to be operational during shutdown margin demonstration when the reactor in Mode 5 with the Mode switch in the Startup/Hot Standby position in accordance with ITS 3.10.8, "SDM Test - Refueling."

8. The SRMs are required to be Operational when in the Refuel mode.

9. The transient analysis discussed in the FSAR does not require the APRMs to be operational in the Refuel Mode to mitigate a transient condition.

The proposed TS changes will not represent a change in the plant as described in the FSAR. FSAR sections 7.5, 12.2A, and 14 were reviewed in making this determination.

In conclusion, monitoring of neutron flux levels, administrative controls, plant procedures, refueling interlocks, and SRM and IRM protective features provide and maintain the defense-in-depth design and operation which precludes the need for the APRMs and APRM Trip Functions to be operable in the Refuel Mode. These changes are consistent with NUREG-1433, Revision 1.

L5 The CTS Table 3.3-1 Note 3.A requirement associated with the Main Steam Isolation Valve Closure Function (ITS Table 3.3.1.1 Function 5), to insert all Operable control rods (MODE 3) within 4 hours, is being relaxed. ITS 3.3.1.1 ACTION F will require that the plant be put in MODE 2 within 6 hours when the Main Steam Isolation Valve Closure Function is inoperable and not restored, or channels tripped, within the required Completion Times. This Function is required only in MODE 1 (current and proposed); therefore, once the plant reaches MODE 2, the LCO is no longer applicable. The current requirement to place the plant in MODE 3 is overly restrictive and inconsistent with CTS LCO 3.0.A. The Main Steam Isolation Valve Closure Function provides protection against over pressure transients in MODE 1, since, with the MSIVs open and the heat generation high, a pressurization transient can occur if the MSIVs close. In Mode 2 the heat generation rate is low enough that other diverse RPS functions provide sufficient protection. The

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L5 (continued)

Completion Time of 6 hours to be in MODE 2 is acceptable due to the low probability of an event requiring this Function during the proposed additional 2 hours. In addition, the 6 hour Completion Time provides sufficient time to reach MODE 2 without challenging plant systems.

L6 The design details in CTS Tables 4.1-1 and 4.1-2 that identify the reliability group (A, B or C) to which each instrument belongs for functional testing, are proposed to be deleted. This design information is not necessary to be included in the Technical Specifications to ensure Operability of these RPS instruments. The requirements in ITS 3.3.1.1 are sufficient to ensure that these RPS instruments are maintained Operable. This change is consistent with NUREG-1433, Revision 1.

L7 The details in CTS Tables 4.1-1, that identify those portions of the instrument channel which require functional testing and the details in CTS Table 4.1-2 that identify the type of test equipment used to perform a channel calibration, are proposed to be deleted. These details are not necessary because the proposed definitions for Channel Functional Test and Channel Calibration provide the necessary guidance. This change is consistent with NUREG-1433, Revision 1.

L8 The details contained in CTS Table 4.1-1, Note 1, concerning testing the automatic scram contactors after maintenance, is proposed to be deleted. Any time the Operability of a system or component has been or could be affected by repair, maintenance, or replacement of a component, post-maintenance testing is required to demonstrate Operability of the system or component. SR 3.0.1 requires the appropriate SRs (in this case, SR 3.3.1.1.4) to be performed to demonstrate Operability of the affected components after work which could affect Operability. Therefore, explicit post maintenance Surveillance Requirements are not required and are proposed to be deleted from the Technical Specifications. Deletion of these details constitutes a less restrictive change. This change is consistent with NUREG-1433, Revision 1.

L9 Not Used.

L10 This change proposes to add a Note (ITS SR 3.3.1.1.3) to the 7 day Channel Functional Test Surveillance Requirement in CTS Table 4.1-1 for the IRM High Flux, IRM Inop, APRM Neutron Flux – High (Startup) Functions. The Note will allow the plant to enter MODE 2 from MODE 1 without performing the required Surveillance. The Surveillance, however, must be performed within 12 hours after entering MODE 2. This is allowed because the testing of the MODE 2 required IRM and APRM

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L10 (continued)

Functions cannot be performed in MODE 1 without utilizing jumpers or lifted leads. Twelve hours is based on operating experience and providing a reasonable time in which to complete the Surveillance Requirement. This change is consistent with NUREG-1433, Revision 1.

L11 The details relating to the Instrument I.D. numbers for the RPS Instrumentation in CTS 4.1.A are proposed to be deleted. These details are not necessary to ensure the RPS instrumentation is maintained Operable. The requirements of ITS 3.3.1.1 (which describes the instrumentation) and the associated Surveillance Requirements are adequate to ensure the required instrumentation is maintained Operable. The Bases also provide a description of the type of instrumentation required by the specification.

L12 This change adds a note to the APRM heat balance calibration of CTS Table 4.1-2 associated with the APRM High Flux output signal (SR 3.3.1.1.2) which states that the Surveillance is not required to be performed until 12 hours after Thermal Power ≥ 25% RTP. This is allowed because it is difficult to accurately determine core Thermal Power from a heat balance when < 25% RTP. Since the APRM Neutron Flux-High (Startup) Function is only required to be Operable in MODE 2 and since the Allowable Value is ≤ 15% RTP, this surveillance is not associated with this Function (ITS 3.3.1.1 Function 2.a). However, the Operability of this Function is assured since an additional surveillance was added to calibrate the entire channel (M11) every 6 months. At low power levels, a high degree of accuracy is unnecessary because of the large inherent margin to power distribution (thermal) limits (MCPR, LHGR, and APLHGR). The 12 hour time limit for performing the surveillance is based on operating experience and providing a reasonable time in which to complete the SR. This change is consistent with NUREG-1433, Revision 1.

L13 The proposed change decreases the Surveillance Frequency for performance of the APRM Heat balance calibration from once per day to once per 7 days. This Surveillance requirement ensures that the APRMs are accurately indicating the true core power which is affected by the LPRM sensitivity. The 7 day Surveillance Frequency is acceptable, based on operating experience and the fact that only minor changes in LPRM sensitivity occur during this time frame. In addition, a review of Surveillance test data during four separate time periods, each in excess of one week, showed that the largest cumulative adjustment was less than 2%. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC) (continued)

L14 The Trip Setting/Trip Level Setting (Allowable Value (A19)) in CTS 2.1.A.3 and CTS Table 3.1-1, Trip Function 15, Turbine Stop Valve Closure is changed from ≤ 10% valve closure to ≤ 15% valve closure (ITS Table 3.3.1.1-1, Function 8, Turbine Stop Valve – Closure) and the Trip Setting/Trip Level Setting (Allowable Value (A19)) in CTS 2.1.A.4 and CTS Table 3.1-1, Trip Function 14, Turbine Control Valve Fast Closure is changed from > 500 psig and < 850 psig to ≥ 500 psig and ≤ 850 psig (ITS Table 3.3.1.1-1, Function 9, Turbine Control Valve Fast Closure, EHC Oil Pressure – Low). The Allowable Values (to be included in the Technical Specifications) and the Trip Setpoints (to be included in plant procedures) have been established consistent with the NYPA Engineering Standards Manual, IES-3A, "Instrument Loop Accuracy and Setpoint Calculation Methodology." The methodology used to determine the "Allowable Values" are consistent with the methodology discussed in ISA-S67.04-1994, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." Any changes to the safety analysis limits, applied in the methodologies, were evaluated and confirmed as ensuring safety analysis licensing acceptance limits are maintained. All design limits, applied in the methodologies, were confirmed as ensuring that applicable design requirements of the associated systems are maintained. The use of this methodology for establishing Allowable Values and Trip Setpoints ensures design or safety analysis limits are not exceeded in the event of transients or accidents and accounts for uncertainties and environmental conditions.

TECHNICAL CHANGES - RELOCATIONS

None