Attachment 3:

- Split Report – Vol. 1
- 3.3 – Vol. 8, 9, and 10

ATTACHMENT 3

REVISION F
SPLIT REPORT AND ITS SECTION 3.3

SUMMARY OF CHANGES TO ITS SECTION 3.3

SUMMARY OF CHANGES TO ITS SECTION 3.3 - REVISION F

Source of Change	Summary of Change	Affected Pages
RAI 3.3.1.1-1	DOC A19 states that the CTS terms "trip level settings" and "trip settings" are the same as the ITS "Allowable Values". The NRC requested that DOC A19 be clarified to better describe why they are the same. DOC A19 has been modified to provide this requested information. In addition, the Bases have been modified to reflect DOC A19 words.	**Specification 3.3.1.1** DOC A19 (DOCs p 5 of 25 and 6 of 25) ITS Bases mark-up p Insert Page B 3.3-3 Retyped ITS Bases P B 3.3-6
RAI 3.3.1.1-2	DOC L4 justified the deletion of the Mode 5 requirements for APRM Neutron Flux - High (Startup) and Inop trips. The DOC based its justifications on a Limerick safety evaluation report. The NRC requested plant specific data to support the change. DOC L4 has been modified to provide the requested data.	**Specification 3.3.1.1** DOC L4 (DOCs p 18 of 25 through 22 of 25) NSHC L4 (NSHCs p 7 of 26 through 10 of 26)
RAI 3.3.1.1-5	The NRC noted that the CTS mark-up for the addition of SR 3.3.1.1.6 (for Function 2.a, the APRM Neutron Flux - High (Startup) trip) incorrectly listed the DOC as L9. This has been corrected to M14.	**Specification 3.3.1.1** CTS mark-up page 14 of 16
RAI 3.3.1.1-6 and TSTF-332	The NRC requested the Response Time Surveillance be modified to reflect TSTF-332. TSTF-332, Rev. 1 modifies the various definitions of response times in Section 1.0, and due to these modifications, the various Notes in the individual Response Time Surveillances are not needed and have been moved to the Bases.	**Specification 3.3.1.1** CTS mark-up page 6 of 16 DOCs A13 (deleted), A14, and LA14 (DOCs p 4 of 25 and 16 of 25) ITS mark-up p 3.3-6 JFDs CLB7 (deleted) and TA1 (JFDs p 1 of 4 and 4 of 4) ITS Bases mark-up p B 3.3-32 and Insert Page B 3.3-32 Bases JFDs CLB3 and TA1 (p 1 of 4 and 3 of 4) Retyped ITS 3.3-6 Retyped ITS Bases B 3.3-35
RAI 3.3.1.1-8	ISTS SR 3.3.1.1.3 requires a Surveillance Test to verify the APRM Neutron Flux - High (Flow Biased) channels conform to a calibrated flow signal every 7 days. The ITS only requires this SR every 92 days, as part of the Channel Functional Test. The NRC requested more justification for this change. JFD CLB2 has been modified to provide a clearer description of the current licensing basis.	**Specification 3.3.1.1** JFD CLB2 (JFDs p 1 of 4)
RAI 3.3.1.1-11	An improper ITS SR reference was used, with respect to the Main Steam Isolation Valve Closure Channel Calibration CTS requirement. This has been corrected.	**Specification 3.3.1.1** CTS mark-up p 14 of 16

Source of Change	Summary of Change	Affected Pages
RAI 3.3.1.1-12	The NRC noted that in the Bases for Function 2.b (APRM Neutron Flux - High (Flow Biased) trip), one paragraph states that no credit is taken for this Function in the safety analyses, yet another paragraph states that credit is taken in the safety analyses for this Function. The Bases have been corrected to clearly state that credit is taken in the thermal-hydraulic instability analysis.	**Specification 3.3.1.1** ITS Bases mark-up p Insert Page B 3.3-8 Retyped ITS Bases p B 3.3-10
RAI 3.3.1.1-13	The NRC noted that the Bases states that the Reactor Pressure - High trip is credited for generator load reject and main turbine trip events when initiated from low power levels, and the trip is required in Modes 1 and 2. However, the Bases also states that at low power levels (e.g., below 29% RTP), the Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, EHC Oil Pressure - Low Functions are not required to be Operable. The NRC believed that the two statements were in conflict with one another and requested clarification. The JFD DB7 has been modified to explain this apparent conflict.	**Specification 3.3.1.1** Bases JFD DB7 (Bases JFDs p 3 of 4)
Amendment 257	This amendment modified a description of what constitutes an acceptable LPRM calibration. However, the change only affects the CTS mark-up (the new Amendment page has been added) since Revision B included the change based on the proposed amendment request.	**Specification 3.3.1.1** CTS mark-up p 14 of 16
Amendment 265	This amendment changed the MSIV Closure Scram trip setting (Allowable Value) from ≤ 10% valve closure from full open to ≤ 15% valve closure from full open.	**Specification 3.3.1.1** CTS mark-up p 4 of 16 and 8 of 16 DOC L9 (deleted) (DOCs p 24 of 25) NSHC L9 (deleted) (NSHC p 17 of 26) ITS mark-up p 3.3-8 Retyped ITS p B 3.3-8
Amendment 266	This amendment affects the Safety Limits Specification (CTS 1.1.A), which is on a CTS mark-up page used by this Specification. However, the Amendment does not affect this Specification.	**Specification 3.3.1.1** CTS mark-up p 1 of 16
TSTF-205	TSTF-205 has been incorporated into the Bases for SR 3.3.1.1.3, SR 3.3.1.1.4, SR 3.3.1.1.8, and SR 3.3.1.1.12, the Channel Functional Tests. The TSTF adds an clarification that, in lieu of testing all the required contacts of a channel relay, only a single contact need be tested (i.e., verify change of state of only a single contact).	**Specification 3.3.1.1** ITS Bases mark-up p B 3.3-27, Insert Page B 3.3-27, B 3.3-29, and Insert Page B 3.3-29 Bases JFD TA2 (JFDs p 3 of 4) Retyped ITS Bases p B 3.3-28, B 3.3-29, and B 3.3-31

Source of Change	Summary of Change	Affected Pages
TSTF-231	TSTF-231 has been incorporated into the Bases for Function 8, the Turbine Stop Valve - Closure Function. The TSTF clarifies that the Function will still cause a trip even if one TSV should fail to close.	Specification 3.3.1.1 ITS Bases mark-up p B 3.3-18 Bases JFD TA3 (Bases JFDs p 4 of 4) Retyped ITS Bases p B 3.3-20
TSTF-355 and WOG-ED-25	TSTF-355 and WOG-ED-25 have been incorporated into the Background Section of the Bases. The TSTF and editorial change clarify the use of the Allowable Value as the Limiting Safety System Setting defined in 10 CFR 50.36.	Specification 3.3.1.1 ITS Bases mark-up p B 3.3-1, Insert Page B 3.3-1a, and Insert Page B 3.3-1b Bases JFD TA4 (Bases JFDs p 4 of 4) Retyped ITS Bases p B 3.3-1, B 3.3-2, and B 3.3-3
New Changes	The Allowable Values for Reactor Pressure - High, Turbine Stop Valve - Closure, and Turbine Control Valve Fast Closure, EHC Oil Pressure - Low Functions have been changed based on recent setpoint calculations.	Specification 3.3.1.1 CTS mark-up p 4 of 16 and 8 of 16 DOCs M16 and L14 (DOCs p 12 of 25 and 25 of 25) NSHC L14 (NSHCs p 25 of 26 and 26 of 26) ITS mark-up p 3.3-8 and 3.3-9 Retyped ITS p 3.3-8 and 3.3-9
Typographical Corrections	Note 2 to SR 3.3.1.1.13 has been modified in the ITS mark-up and retyped ITS to reflect the proper Function number (Function 1.a). A typographical correction has been made to the retyped ITS Table 3.3.1.1-1 (the wrong SR number was identified for Function 1.b). A change in the CTS mark-up was annotated with an incorrect DOC number (L9 was replaced with M5).	Specification 3.3.1.1 CTS mark-up p 12 of 16 ITS mark-up p 3.3-5 Retyped ITS p 3.3-5 and 3.3-7
RAI 3.3.1.2-1	The NRC noted that a DOC was not provided to justify the addition of the signal to noise ratio verification, nor was the change annotated in the CTS mark-up. This has been corrected.	Specification 3.3.1.2 CTS mark-up p 2 of 3 DOC M7 (DOCs p 3 of 7) ITS mark-up p 3.3-13
RAI 3.3.1.2-2	The NRC requested that an enhanced safety basis discussion be added to DOC L1, which justified adding Actions for when one or more SRMs are inoperable.	Specification 3.3.1.2 DOC L1 (DOCs p 5 of 7)

SUMMARY OF CHANGES TO ITS SECTION 3.3 - REVISION F

Source of Change	Summary of Change	Affected Pages
TSTF-205	TSTF-205 has been incorporated into the Bases for SR 3.3.1.2.5 and SR 3.3.1.2.6, the Channel Functional Tests. The TSTF adds an clarification that, in lieu of testing all the required contacts of a channel relay, only a single contact need be tested (i.e., verify change of state of only a single contact).	Specification 3.3.1.2 ITS Bases mark-up p B 3.3-41 and Insert Page B 3.3-41 JFD TA1 (JFDs p 1 of 1) Retyped ITS Bases p B 3.3-45
RAI 3.3.2.1-2	The NRC noted that the CTS mark-up showed the addition of SR 3.3.2.1.1 (a Channel Functional Test) for Function 1.b, RBM - Inop, but did not describe the addition in DOC M1. This oversight has been corrected.	Specification 3.3.2.1 DOC M1 (DOCs p 2 of 9)
RAI 3.3.2.1-3 (as modified)	The NRC requested more detailed information justifying the change in the RWM Channel Functional Test Frequency. This has been provided.	Specification 3.3.2.1 DOC L3 (DOCs p 6 of 9 and 7 of 9) NSHC L3 (NSHCs p 5 of 15 and 6 of 15) JFD DB2 (JFDs p 1 of 3)
RAI 3.3.2.1-4	The NRC questioned relocating a CTS reporting requirement (required when a startup is performed with the RWM inoperable) to the Bases. Therefore, the CTS requirement will not be added to the Bases, but will be deleted.	Specification 3.3.2.1 CTS mark-up p 9 of 10 DOCs LA4 (deleted) and L8 (DOCs p 4 of 9 and 8 of 9 and 9 of 9) NSHC L8 (NSHCs p 14 of 15 and 15 of 15) ITS Bases mark-up p B 3.3-50 Retyped ITS Bases p B 3.3-53
RAI 3.3.2.1-5	The NRC requested additional information to discuss the safety basis for not requiring a daily channel check of the RBM - Upscale and RBM - Downscale Functions. This has been provided.	Specification 3.3.2.1 DOC L2 (DOCs p 6 of 9)
RAI 3.3.2.1-6	CTS 4.3.B.5 requires an RBM functional test prior to withdrawal of control rods when a limiting control rod pattern exists. However, the NRC noted that DOC L6 justifies deleting this testing requirement because performing a functional test due to one channel being inoperable does not increase the reliability of the other channel. The NRC requested that the DOC be corrected to reflect the proper reason for deleting the test. This has been provided.	Specification 3.3.2.1 DOC L6 (DOCs p 7 of 9 and 8 of 9) NSHC L6 (NSHCs p 10 of 15 and 11 of 15)

Source of Change	Summary of Change	Affected Pages
TSTF-205	TSTF-205 has been incorporated into the Bases for SR 3.3.2.1.1, SR 3.3.2.1.2, SR 3.3.2.1.3, and SR 3.3.2.1.7, the Channel Functional Tests. The TSTF adds an clarification that, in lieu of testing all the required contacts of a channel relay, only a single contact need be tested (i.e., verify change of state of only a single contact).	<u>Specification 3.3.2.1</u> ITS Bases mark-up p B 3.3-51, Insert Page B 3.3-51, B 3.3-53, and Insert Page B 3.3-53 Bases JFD TA1 (Bases JFDs p 2 of 2) Retyped ITS Bases p B 3.3-55, B 3.3-56, and B 3.3-57
NRC-ED-14	An editorial generic change has been made. NRC-ED-14 modified Required Action D.1 to be consistent with Required Action C.2.2 (the word "accordance" was changed to "compliance").	<u>Specification 3.3.2.1</u> ITS mark-up p 3.3-16 JFD TA1 (JFDs p 3 of 3) Retyped ITS p 3.3-16
New Change	The Applicable Safety Analyses section of the Bases have been modified, consistent with the changes made for RAI 3.3.1.1-1.	<u>Specification 3.3.2.1</u> ITS Bases mark-up p Insert Page B 3.3-46 Retyped ITS Bases p B 3.3-50
Typographical correction	The RBM - Downscale Allowable Value should be "2.5/125" divisions of full scale, not "2.5" divisions of full scale.	<u>Specification 3.3.2.1</u> ITS mark-up p 3.3-20 Retyped ITS p 3.3-20
RAI 3.3.2.2-1	DOC A6 states that the CTS term "trip level settings" is the same as the ITS "Allowable Values". The NRC requested that DOC A6 be clarified to better describe why they are the same. DOC A6 has been modified to provide this requested information. In addition, the Bases have been modified to reflect DOC A6 words.	<u>Specification 3.3.2.2</u> DOC A6 (p 2 of 6 and 3 of 6) ITS Bases mark-up p Insert Page B 3.3-57 Retyped ITS Bases p B 3.3-61 and B 3.3-62
TSTF-205	TSTF-205 has been incorporated into the Bases for SR 3.3.2.2.2, the Channel Functional Test. The TSTF adds an clarification that, in lieu of testing all the required contacts of a channel relay, only a single contact need be tested (i.e., verify change of state of only a single contact).	<u>Specification 3.3.2.2</u> ITS Bases mark-up p B 3.3-61 and Insert Page B 3.3-61 Bases JFD TA2 (Bases JFDs p 2 of 2) Retyped ITS Bases p B 3.3-65

SUMMARY OF CHANGES TO ITS SECTION 3.3 - REVISION F

Source of Change	Summary of Change	Affected Pages
TSTF-297 and BWROG-ED-7	TSTF-297 adds an allowance in lieu of shutting down the plant when the inoperable channels are not tripped or restored within the Completion Time of Required Actions A.1 or B.1. The TSTF will allow the affected stop valve(s) to be removed from service if the inoperable channels is the result of an inoperable stop valve. Generic editorial change BWROG-ED-7 added "(s)" to the TSTF-297 Bases changes in two locations for correctness.	<u>Specification 3.3.2.2</u> CTS mark-up p 2 of 3 DOC L2 (DOCs p 5 of 6 and 6 of 6) NSHC L2 (NSHCs p 3 of 4 and 4 of 4) ITS mark-up p 3.3-21 JFDs PA1, DB6, and TA1 (JFDs p 1 of 2 and 2 of 2) ITS Bases mark-up p B 3.3-60 and Insert Page B 3.3-60 Bases JFDs DB5, TA1, and TA3 (Bases JFDs p 1 of 2 and 2 of 2) Retyped ITS p 3.3-21 Retyped ITS Bases p B 3.3-64
New Change	The Allowable Value for the Reactor Vessel Water Level - High has been changed from "222.5" inches to "222.4" inches, based on a recent setpoint calculation.	<u>Specification 3.3.2.2</u> CTS mark-up p 2 of 3 DOC M.1 (DOCs p 1 of 1) ITS mark-up p 3.3-22 Retyped ITS p 3.3-22
RAI 3.3.3.1-1 (as modified) and RAI 3.3.3.1-2 (as modified)	CTS Table 3.2-8, Note K allows the Primary Containment H_2 and O_2 Concentration Monitors to be inoperable for 3 hours per 24 hour period when the PASS is being operated. Portions of this allowance was moved to the Bases by DOC LA3 and portions were deleted by DOC L5. The NRC noted that the Bases could not be used to change the Technical Specifications and requested that this be moved back to the Technical Specifications. The allowance has been placed back into the Technical Specifications as a Note to the LCO statement.	<u>Specification 3.3.3.1</u> CTS mark-up p 5 of 7 DOCs LA3 (deleted) and L5 (deleted) (DOCs p 5 of 8 and 6 of 8) NSHC L5 (deleted) (NSHCs p 8 of 10) ITS mark-up p 3.3-23 JFD CLB5 (JFDs p 1 of 3) ITS Bases mark-up p Insert Page B 3.3-68 Retyped ITS p 3.3-23 Retyped ITS Bases p B 3.3-74

Source of Change	Summary of Change	Affected Pages
RAI 3.3.3.1-3 and TSTF-295	Note 2 to the ACTIONS was modified to allow separate Condition entry on a per penetration flow path basis for Function 7, the PCIV Position Function. The NRC requested that TSTF-295 be included in this Specification. TSTF-295 modifies the PCIV Position Function in ITS Table 3.3.3.1-1 to be on a penetration flow path basis; thus the modified Note is not necessary.	<u>Specification 3.3.3.1</u> CTS mark-up p 4 of 7 DOC M3 (DOCs p 3 of 8) ITS mark-up p 3.3-23 and 3.3-26 JFDs TA1 and X1 (deleted) (JFDs p 2 of 3) ITS Bases mark-up p B 3.3-67 Bases JFD TA1 (Bases JFDs p 2 of 3) Retyped ITS p 3.3-23 and 3.3-26 Retyped ITS Bases p B 3.3-73
RAI 3.3.3.1-4 (as modified)	The NRC requested that DOC M1 be clarified to clearly state that ITS ACTIONS A and C are new requirements, since the ITS now requires two channels to be Operable where the CTS only requires one channel to be Operable. DOC M1 has been modified to provide this clarification.	<u>Specification 3.3.3.1</u> DOC M1 (DOCs p 3 of 8) ITS mark-up p 3.3-23
RAI 3.3.3.1-5	The NRC requested that a safety basis justification be provided to justify the relocation of certain details to the Bases (the details concerning the remedial actions to perform alternate sampling and analysis for inoperable PAM channels during the 30 day allowed outage time). DOC LA2 has been modified accordingly.	<u>Specification 3.3.3.1</u> DOCs LA2 (p 4 of 8)
RAI 3.3.3.1-7 (as modified)	The NRC noted that the addition of Table 3.3.3.1-1 Footnotes (a) and (b) were discussed in the DOCs and requested proper justification for these footnotes. DOC M3 has been revised accordingly.	<u>Specification 3.3.3.1</u> DOC M3 (DOCs p 3 of 8)
RAI 3.3.3.1-8 (as modified)	The NRC noted that the Retyped ITS included Table 3.3.3.1-1 Footnote (c), whereas the ITS mark-up, CTS mark-up, and DOCs did not show this footnote. The NRC requested that proper justification be provided for the addition of this footnote. The Retyped ITS was in error, in that the footnote should not be included in the ITS. Thus, it has been deleted.	<u>Specification 3.3.3.1</u> ITS mark-up p 3.3-26 JFDs CLB6 and TA1 (JFDs p 1 of 3 and 2 of 3) ITS Bases mark-up p B 3.3-69 Bases JFDs CLB5 and TA1 (Bases JFDs p 1 of 3 and 2 of 3) Retyped ITS p 3.3-26

Source of Change	Summary of Change	Affected Pages
RAI 3.3.3.1-9 and BSI-20	The NRC noted in BSI-20 that CTS Functions 15 through 18 (core spray flow, core spray discharge pressure, RHR flow, and RHRSW flow) do not appear to be Type A nor Category 1 instruments. The NRC stated that if the licensee determined that these Functions are not Type A, Category 1 instruments, then they would entertain a proposal to remove the instruments from the ITS. JAFNPP has determined that these instruments are not Type A not Category 1 instruments, therefore a new R DOC has been written to relocate the instruments to the TRM.	**Specification 3.3.3.1** CTS mark-up p 4 of 7, 5 of 7, and 7 of 7 DOCs L7 (deleted) and R2 (DOCs p 6 of 8, 7 of 8, and 8 of 8) NSHC L7 (deleted) (NSHCs p 10 of 10) ITS mark-up p 3.3-26 ITS Bases mark-up p B 3.3-64, B 3.3-65, Insert Page B 3.3-65, B 3.3-69, and Insert Page B 3.3-69 Bases JFD CLB1 (Bases JFDs p 1 of 3) Retyped ITS p 3.3-26 Retyped ITS Bases p B 3.3-69 and B 3.3-75 **Split Report** Appendix A p 5 of 23 and 6 of 23
RAI 3.3.3.1-10 and BSI-21	The ITS included the Refueling Zone Water Level instrument, which is not in the CTS. The NRC noted that this instrument is a Category 3 instrument, and did not seem appropriate for inclusion in the ITS. This channel has been deleted from the ITS, consistent with the CTS.	**Specification 3.3.3.1** CTS mark-up p 3 of 7, 5 of 7, and 6 of 7 DOC M4 (DOCs p 4 of 8) ITS mark-up p 3.3-26 ITS Bases mark-up p B 3.3-64, B 3.3-65, Insert Page B 3.3-65, and B 3.3-66 Bases JFD CLB1 (Bases JFDs p 1 of 3) Retyped ITS p 3.3-26 Retyped ITS Bases p B 3.3-69, B 3.3-70, and B 3.3-71
New Change	A change has been made to the Suppression Pool Water Temperature Function Bases due to a recent UFSAR change.	**Specification 3.3.3.1** ITS Bases mark-up p Insert Page B 3.3-69 Retyped ITS Bases p B 3.3-75

Source of Change	Summary of Change	Affected Pages
RAI 3.3.3.2-1 (as modified)	CTS 3.2.J.3.a provides an option to place the component actuated by the control circuit in the safe shutdown configuration. The ITS deleted this option with an Administrative DOC and the NRC requested more information to conclude that this deletion was administrative. This deletion is actually more restrictive, therefore the A DOC has been deleted and a new M DOC has been provided.	<u>Specification 3.3.3.2</u> CTS mark-up p 1 of 11 DOCs A3 (deleted) and M3 (DOCs p 1 of 4, 2 of 4, and 3 of 4)
RAI 3.3.3.2-2 (as modified)	CTS 3.2.J.2.b provides an allowance to establish an alternate method of monitoring the parameter within 30 days and to restore the instrument to Operable status within 90 days. The ITS deleted portions of this requirement with a less restrictive DOC and relocated portions to the Bases with an LA DOC. The NRC Noted that this did not appear correct, specifically the relocated portions. The ITS has been modified to delete the entire option with a more restrictive DOC.	<u>Specification 3.3.3.2</u> CTS mark-up p 1 of 11 DOCs M3, LA1 (deleted) and L1 (deleted) (DOCs p 2 of 4 and 3 of 4)
RAI 3.3.3.2-3 (as modified)	The NRC noted that the ITS relocated the list of Remote Shutdown instrumentation to the TRM, while TSTF-266 relocated the instrumentation to the Bases. The NRC requested that the ITS be modified to reflect the TSTF. However, the NRC has allowed many plants (e.g., WNP-2, NMP2, and LaSalle County Station Units 1 and 2) to relocate this instrumentation to the TRM. Therefore, the ITS Bases will not include the instrumentation list and DOC LA2 and JFD X2 have been modified to reflect recent plants that have been approved similarly.	<u>Specification 3.3.3.2</u> DOC LA1 (DOCs p 3 of 4) JFD X2 (JFDs p 1 of 1) Bases JFD X2 (Bases JFDs p 1 of 1)
TSTF-367	TSTF-367 has been incorporated into the Applicable Safety Analyses section of the Bases. The TSTF revises the Bases to reference Criterion 4 of 10 CFR 50.36(c)(2)(ii).	<u>Specification 3.3.3.2</u> ITS Bases mark-up p B 3.3-75 Bases JFD TA1 (Bases JFDs p 1 of 1) Retyped ITS Bases p B 3.3-81

Source of Change	Summary of Change	Affected Pages
RAI 3.3.4.1-1	The NRC noted that the ITS and Bases discussion of the channel and trip system description were inconsistent and requested appropriate corrections. The logic description has been corrected, consistent with the design.	Specification 3.3.4.1 CTS mark-up p 2 of 6 and 3 of 6 DOCs A3 (deleted), A6, and L1 (DOCs p 1 of 8 and 5 of 8) ITS mark-up p 3.3-33 JFD DB1 (deleted) (JFDs p 1 of 2 ITS Bases mark-up p B 3.3-91, Insert Page B 3.3-91, B 3.3-92, B 3.3-93, B 3.3-94, B 3.3-95, and B 3.3-96 Bases JFD CLB3 and DB1 (deleted) (Bases JFDs p 1 of 2) Retyped ITS p 3.3-29 Retyped ITS Bases p B 3.3-86, B 3.3-89, B 3.3-90, and B 3.3-91
RAI 3.3.4.1-2 (as modified)	CTS footnote allows an inoperable instrument to not be placed in trip when this would cause an actual trip to occur. The ITS added a new Required Action to restore the instrument to Operable status, and DOC A4 stated that this new Required Action was equivalent to the CTS footnote. The NRC requested that this comparison be clarified. The new Required Action is actually a less restrictive change and the deletion of the CTS footnote is an LA change (relocated to the Bases). These corrections have been made.	Specification 3.3.4.1 CTS mark-up p 3 of 6 DOCs A4 (deleted), LA2, and L1 (DOCs p 1 of 8, 5 of 8 and 6 of 8)
RAI 3.3.4.1-3	ITS Required Action A.2 Note has been added to restrict use of the Required Action to trip a channel if the inoperable channel is the result of an inoperable breaker. This addition was justified by an administrative DOC. The NRC requested clarification to justify why this change was administrative. The change is actually a more restrictive change, thus a new M DOC has been provided.	Specification 3.3.4.1 CTS mark-up p 3 of 6 DOCs A5 (deleted) and M3 (DOCs p 1 of 8 and 4 of 8)
RAI 3.3.4.1-4	DOC A7 states that the CTS term "trip level settings" is the same as the ITS "Allowable Values". The NRC requested that DOC A7 be clarified to better describe why they are the same. DOC A7 has been modified to provide this requested information. In addition, the Bases have been modified to reflect DOC A7 words.	Specification 3.3.4.1 DOC A7 (DOCs p 1 of 8 and 2 of 8) ITS Bases mark-up p Insert Page B 3.3-92 Retyped ITS Bases p B 3.3-87 and B 3.3-88

Source of Change	Summary of Change	Affected Pages
RAI 3.3.4.1-5	Certain changes to the CTS allowed outage times were justified by GENE-770-06-1-A. The NRC requested a license amendment citation for this analysis. The DOCs have been revised appropriately.	**Specification 3.3.4.1** DOCs L1 and L2 (DOCs p 6 of 8, 7 of 8, and 8 of 8)
RAI 3.3.4.1-6	When referring to the ATWS Reactor Pressure High setpoint, the CTS bases it upon the number of inoperable S/RVs while the ITS bases it upon the number of OPERABLE S/RVs. The NRC requested a specific DOC justifying the change. This specific DOC has been provided.	**Specification 3.3.4.1** CTS mark-up p 4 of 6 DOC A9 (DOCs p 3 of 8) ITS mark-up p 3.3-35
TSTF-205	TSTF-205 has been incorporated into the Bases for SR 3.3.4.1.2, the Channel Functional Test. The TSTF adds an clarification that, in lieu of testing all the required contacts of a channel relay, only a single contact need be tested (i.e., verify change of state of only a single contact).	**Specification 3.3.4.1** ITS Bases mark-up p B 3.3-98 and Insert Page B 3.3-98 JFD TA1 (JFDs p 2 of 2) Retyped ITS Bases p B 3.3-94
TSTF-297 and BWROG-ED-7	TSTF-297 has been incorporated into Required Action D.1. The TSTF adds a Note restricting the use of Required Action D.1 (which removes the affected recirculation pump from service) to when the inoperable channel is the result of an inoperable RPT breaker only. In addition, BWROG-ED-7 provided a clarification to the Bases changes of TSTF-297.	**Specification 3.3.4.1** DOC L3 (DOCs p 8 of 8) ITS mark-up p 3.3-34 JFD TA1 (JFDs p 1 of 2) ITS Bases mark-up p B 3.3-97 and Insert Page B 3.3-97 Bases JFDs TA2 and TA3 (Bases JFDs p 2 of 2) Retyped ITS p 3.3-30 Retyped ITS Bases p B 3.3-92
TSTF-367	TSTF-367 has been incorporated into the Applicable Safety Analyses section of the Bases. The TSTF revises the Bases to reference Criterion 4 of 10 CFR 50.36(c)(2)(ii).	**Specification 3.3.4.1** ITS Bases mark-up p B 3.3-92 Bases JFD TA4 (Bases JFDs p 2 of 2) Retyped ITS Bases p B 3.3-87
Amendment 237	A blank CTS mark-up page was inadvertently not updated with the proper amendment number after the amendment was approved. The proper blank CTS mark-up page has been provided.	**Specification 3.3.4.1** CTS mark-up p 6 of 6

Source of Change	Summary of Change	Affected Pages
Amendment 264	Amendment 264, which modified the Reactor Water Level - Low Low Allowable Value, has been incorporated into the ITS.	<u>Specification 3.3.4.1</u> CTS mark-up p 2 of 6 ITS mark-up p 3.3-35 ITS Bases mark-up p Insert Page B 3.3-94 Retyped ITS p 3.3-31 Retyped ITS Bases p B 3.3-89
New Change	The Allowable Values for the Reactor Pressure - High Function have been changed based on recent setpoint calculations.	<u>Specification 3.3.4.1</u> CTS mark-up p 2 of 6 DOC M2 (DOCs p 4 of 8) ITS mark-up p 3.3-35 JFDs DB3 and DB4 (JFDs p 1 of 2) Retyped ITS p 3.3-31
RAI 3.3.5.1-1	DOC A12 states that the CTS term "trip level settings" is the same as the ITS "Allowable Values". The NRC requested that DOC A12 be clarified to better describe why they are the same. DOC A12 has been modified to provide this requested information. In addition, the Bases have been modified to reflect DOC A12 words.	<u>Specification 3.3.5.1</u> DOC A12 (DOCs p 5 of 14) ITS Bases mark-up p Insert Page B 3.3-108 Retyped ITS Bases p B 3.3-105
TSTF-205	TSTF-205 has been incorporated into the Bases for SR 3.3.5.1.2, the Channel Functional Test. The TSTF adds an clarification that, in lieu of testing all the required contacts of a channel relay, only a single contact need be tested (i.e., verify change of state of only a single contact).	<u>Specification 3.3.5.1</u> ITS Bases mark-up p B 3.3-136 and Insert Page B 3.3-136 Bases JFD TA1 (Bases JFDs p 3 of 4) Retyped ITS Bases p B 3.3-133

Source of Change	Summary of Change	Affected Pages
TSTF-275	TSTF-275 has been incorporated into Table 3.3.5.1-1 footnote (a). The TSTF clarifies that the Reactor Vessel Water Level - Low Low Low (Level 1) and Drywell Pressure - High Functions are only required when the associated ECCS subsystems are required to be Operable.	Specification 3.3.5.1 DOC A10 (DOCs p 4 of 14) ITS mark-up p 3.3-42, 3.3-43, and 3.3-44 JFD TA1 (JFDs p 3 of 4) ITS Bases mark-up p B 3.3-107, Insert Page B 3.3-107, B 3.3-108, Insert Page B 3.3-108, B 3.3-109, Insert Page B 3.3-109, B 3.3-110, Insert Page B 3.3-110, B 3.3-111, Insert Page B 3.3-111a, and Insert Page B 3.3-111b Bases JFD TA2 (Bases JFDs p 4 of 4) Retyped ITS p 3.3-38, 3.3-39, and 3.3-40 Retyped ITS Bases p B 3.3-105, B 3.3-107, B 3.3-108, B 3.3-109, and B 3.3-110
Amendment 263	This Amendment modified the Allowable Values for the RHR and CS pump start timers and ADS auto blowdown timers and extended the Frequencies for the Channel Calibration and LSFT Surveillances.	Specification 3.3.5.1 CTS mark-up p 3 of 15, 4 of 15, 10 of 15, and 11 of 15 DOCs L1 (deleted) and L5 (deleted) (DOCs p 11 of 14 and 13 of 14) NSHC L1 (deleted) and L5 (deleted) (NSHCs p 1 of 13 and 8 of 13) ITS mark-up p 3.3-41 JFDs CLB4, DB8, and X1 (JFDs p 1 of 4, 3 of 4, and 4 of 4) ITS Bases mark-up P B 3.3-138 Bases JFD X2 (deleted) (Bases JFDs p 4 of 4)

SUMMARY OF CHANGES TO ITS SECTION 3.3 - REVISION F

Source of Change	Summary of Change	Affected Pages
New Changes	The Allowable Values for the CS Pump Discharge Flow - Low (Bypass), LPCI Reactor Pressure - Low (Recirculation Discharge Valve Permissive), LPCI Pump Discharge Flow - Low (Bypass), LPCI Containment Pressure - High, HPCI Reactor Vessel Water Level - High (Level 8), and HPCI Pump Discharge Flow - Low (Bypass), Functions have been changed based on recent setpoint calculations. In addition, for consistency with other modified DOCs, the Revision of the Engineering Standards Manual has been deleted from DOC M6.	<u>Specification 3.3.5.1</u> CTS mark-up p 2 of 15 and 5 of 15 DOCs M6, M7 and L6 (DOCs p 9 of 14 and 13 of 14) NSHC L6 (NSHCs p 9 of 13, 10 of 13, and 11 of 13) ITS mark-up p 3.3-42, 3.3-43, 3.3-44, and 3.3-45 Retyped ITS p 3.3-38, 3.3-39, 3.3-40, and 3.3-41
New Change	The minimum Allowable Value for the CS and LPCI pump start timers have been deleted from this Specification. The safety analysis only assumes the ECCS pumps start within a maximum time; the minimum time only affects the EDG, and this requirement is already covered by a EDG Surveillance.	<u>Specification 3.3.5.1</u> CTS mark-up p 3 of 15 DOC L7 (DOCs p 14 of 14) NSHC L7 (NSHCs p 12 of 13 and 13 of 13) ITS mark-up p Insert Page 3.3-42 and B 3.3-43 ITS Bases mark-up p Insert Page B 3.3-111a Retyped ITS p 3.3-38 and 3.3-39 Retyped ITS Bases p B 3.3-109
New Change	The Bases for the Reactor Water Level - Low Low, Level 2 have been modified to clarify that the Allowable Values for all the Low Low, Level 2 Functions (i.e., HPCI, RCIC, and ATWS-RPT) are not the same.	<u>Specification 3.3.5.1</u> ITS Bases mark-up p Insert Page B 3.3-115 Retyped ITS Bases p B 3.3-114
Typographical Correction	The Allowable Value for ITS Table 3.3.5.1-1, Function 2.e in the Retyped ITS has been changed to match the ITS mark-up.	<u>Specification 3.3.5.1</u> Retyped ITS p 3.3-40
RAI 3.3.5.1-1	DOC A9 states that the CTS term "trip level settings" is the same as the ITS "Allowable Values". The NRC requested that DOC A9 be clarified to better describe why they are the same. DOC A9 has been modified to provide this requested information. In addition, the Bases have been modified to reflect DOC A9 words.	<u>Specification 3.3.5.2</u> DOC A9 (DOCs p 3 of 7 and 4 of 7) ITS Bases mark-up p Insert Page B 3.3-141 Retyped ITS Bases p B 3.3-138

SUMMARY OF CHANGES TO ITS SECTION 3.3 - REVISION F

Source of Change	Summary of Change	Affected Pages
TSTF-205	TSTF-205 has been incorporated into the Bases for SR 3.3.5.2.2, the Channel Functional Test. The TSTF adds an clarification that, in lieu of testing all the required contacts of a channel relay, only a single contact need be tested (i.e., verify change of state of only a single contact).	Specification 3.3.5.2 ITS Bases mark-up p B 3.3-149 and Insert Page B 3.3-149 JFD TA1 (JFDs p 2 of 2) Retyped ITS Bases p B 3.3-146
TSTF-367	TSTF-367 has been incorporated into the Applicable Safety Analyses section of the Bases. The TSTF revises the Bases to reference Criterion 4 of 10 CFR 50.36(c)(2)(ii).	Specification 3.3.5.2 ITS Bases mark-up p B 3.3-140 Bases JFD TA2 (Bases JFDs p 2 of 2) Retyped ITS Bases p B 3.3-137
Amendment 263	This amendment affects the ECCS Instrumentation Specification (CTS 3.2.B), which is on a CTS mark-up page used by this Specification. However, the amendment does not affect this Specification.	Specification 3.3.5.2 CTS mark-up p 3 of 10 and 8 of 10
New Change	The Allowable Value for the RCIC Reactor Vessel Water Level - High (Level 8) Function has been changed based on a recent setpoint calculation.	Specification 3.3.5.2 CTS mark-up p 2 of 10 DOC M4 (DOCs p 5 of 7) ITS mark-up p 3.3-51 Retyped ITS p 3.3-46
New Change	The Bases for the Reactor Water Level - Low Low, Level 2 have been modified to clarify that the Allowable Values for all the Low Low, Level 2 Functions (i.e., HPCI, RCIC, and ATWS-RPT) are not the same.	Specification 3.3.5.2 ITS Bases mark-up p Insert Page B 3.3-141 Retyped ITS Bases p B 3.3-139
RAI 3.3.6.1-2	DOC A16 states that the CTS term "trip level settings" is the same as the ITS "Allowable Values". The NRC requested that DOC A16 be clarified to better describe why they are the same. DOC A16 has been modified to provide this requested information. In addition, the Bases have been modified to reflect DOC A16 words.	Specification 3.3.6.1 DOC A16 (DOCs p 4 of 26 and 5 of 26) ITS Bases mark-up p Insert Page B 3.3-156 Retyped ITS Bases p B 3.3-155

Source of Change	Summary of Change	Affected Pages
RAI 3.3.6.1-3	The CTS requires a plant shutdown when the RHR Shutdown Cooling Isolation on Reactor Vessel Water Level - Low (Level 3) is not restored with the allowed completion time. The ITS only requires action to be initiated to either restore the channels or isolate the RHR Shutdown Cooling System. The NRC requested further justification concerning how this change is consistent with the plant safety analysis. DOC L5 has been modified to provide this information.	**Specification 3.3.6.1** CTS mark-up p 4 of 25 DOC L5 (DOCs p 19 of 26 and 20 of 26)
RAI 3.3.6.1-4	CTS Table 3.2-1 Action 3.B requires the main steam lines to be isolated within 8 hours when the Main Steam Line Pressure - Low channels are not restored within the allowed completion time. The ITS requires placing the plant in MODE 2 within 8 hours. The NRC noted that the NUREG normally allows 6 hours to place the plant in MODE 2, and requested additional justification for the 8 hour time.	**Specification 3.3.6.1** DOC L15 (DOCs p 24 of 26)
RAI 3.3.6.1-5	The CTS lists the trip level setting for the RWCU, HPCI, and RCIC Area Temperature Functions as "≤ 40°F above max. ambient." In the ITS, the Allowable Values are specific temperature values. The NRC noted that the change was described as an administrative change, and requested additional justification to verify these changes are administrative. JAFNPP has determined that these changes should have been classified as more restrictive. Therefore a new M DOC has been provided for these changes.	**Specification 3.3.6.1** CTS mark-up p 3 of 25 and 4 of 25 DOCs A7 and M14 (DOCs p 2 of 26, 11 of 26, and 12 of 26) ITS mark-up p 3.3-57, 3.3-59, Insert Page 3.3-59, 3.3-60, Insert Page 3.3-60, 3.3-61, and Insert Page 3.3-61
RAI 3.3.6.1-9	The ITS did not include the term "automatic" in Condition B, since all primary containment isolation Functions in the JAFNPP ITS are automatic. The NRC requested additional clarification be provided for this change. JFD CLB7 and Bases JFD CLB1 have been modified to clarify the change.	**Specification 3.3.6.1** JFD CLB7 (JFDs p 2 of 5) Bases JFD CLB1 (Bases JFDs p 1 of 4)
Amendment 257	This amendment affects the RPS Instrumentation Specification (CTS 3.1.A),.which is on a CTS mark-up page used by this Specification. However, the amendment does not affect this Specification; it only results in the renumbering of the CTS mark-up pages.	**Specification 3.3.6.1** CTS mark-up p 1 of 25 through 25 of 25
Amendment 263	This amendment affects the ECCS Instrumentation Specification (CTS 3.2.B), which is on a CTS mark-up page used by this Specification. However, the amendment does not affect this Specification.	**Specification 3.3.6.1** CTS mark-up p 18 of 25 and 21 of 25
Amendment 265	This amendment affects the RPS Instrumentation Specification, which is on a CTS mark-up page used by this Specification. However, the amendment does not affect this Specification.	**Specification 3.3.6.1** CTS mark-up p 22 of 25

Source of Change	Summary of Change	Affected Pages
TSTF-205	TSTF-205 has been incorporated into the Bases for SR 3.3.6.1.2, the Channel Functional Test. The TSTF adds an clarification that, in lieu of testing all the required contacts of a channel relay, only a single contact need be tested (i.e., verify change of state of only a single contact).	**Specification 3.3.6.1** ITS Bases mark-up p B 3.3-181 and Insert Page B 3.3-181 Bases JFD TA1 (Bases JFDs p 4 of 4) Retyped ITS Bases p B 3.3-181
TSTF-306	TSTF-306 has been incorporated. This TSTF adds a Note to the ACTIONS to allow penetration flow paths, closed to comply with the ACTIONS, to be unisolated intermittently under administrative control. The TSTF also adds a new ACTION for the TIP System Isolation Valves.	**Specification 3.3.6.1** CTS mark-up p 3 of 25, 5 of 25, 6 of 25, 11 of 25, 13 of 25, 16 of 25, and 23 of 25 DOCs A5, A15, M2, M7, M11, L5, L11, L18, and L19 (DOCs p 2 of 26, 4 of 26, 6 of 26, 8 of 26, 10 of 26, 19 of 26, 22 of 26, 25 of 26, and 26 of 26) NSHCs L11, L18, and L19 (NSHCs p 16 of 32, 17 of 32, 29 of 32, 30 of 32, 31 of 32, and 32 of 32) ITS mark-up p 3.3-52, 3.3-53, 3.3-54, 3.3-55, 3.3-58, 3.3-61, and 3.3-62 JFDs CLB1, CLB 7, and TA1 (JFDs p 1 of 5, 2 of 5, and 5 of 5) ITS Bases mark-up p B 3.3-155, Insert Page B 3.3-155, B 3.3-174, Insert Page B 3.3-174a, Insert Page B 3.3-174b, B 3.3-175, Insert Page B 3.3-175, Insert Page B 3.3-176, B 3.3-178, B 3.3-179, Insert Page B 3.3-179, B 3.3-180, and Insert Page B 3.3-180 Bases JFDs CLB1 and TA3 (Bases JFDS p 1 of 4, and 4 of 4) Retyped ITS p 3.3-47, 3.3-48, 3.3-49, 3.3-50, 3.3-53, 3.3-56, and 3.3-57 Retyped ITS Bases p B 3.3-154, B 3.3-173, B 3.3-174, B 3.3-175, B 3.3-176, B 3.3-178, B 3.3-179, and B 3.3-180

Source of Change	Summary of Change	Affected Pages
TSTF-332	TSTF-332, Rev. 1 modifies the various definitions of response times in Section 1.0, and due to these modifications, the various Notes in the individual Response Time Surveillances are not needed and have been moved to the Bases.	**Specification 3.3.6.1** CTS mark-up p 2 of 25 DOCs A14 and LA12 (DOCs p 4 of 26, 17 of 26, and 18 of 26) ITS mark-up p 3.3-56 JFDs CLB6 (deleted) and CLB8 (JFDs p 1 of 5 and 2 of 5) ITS Bases mark-up p B 3.3-183 and Insert Page B 3.3-183 Bases JFDs CLB4 and TA2 (Bases JFDs p 1 of 4 and 4 of 4) Retyped ITS p 3.3-51 Retyped ITS Bases p B 3.3-184
New Changes	The Allowable Values for the Main Steam Line Flow - High, Main Steam Tunnel Area Temperature - High, HPCI Steam Line Flow - High, HPCI Turbine Exhaust Diaphragm Pressure - High, HPCI Steam Line Penetration (Drywell Entrance) Area Temperature - High, HPCI Steam Line Torus Room Area Temperature - High, HPCI Equipment Area Temperature - High, RHR Heat Exchanger A Area Temperature - High, RHR Heat Exchanger B Area Temperature - High, RB Southwest Area of Elevation 272' Temperature - High, RB Southeast Area of Elevation 272' Temperature - High, RCIC Steam Line Flow - High, RCIC Steam Supply Line Pressure - Low, RCIC Turbine Exhaust Diaphragm Pressure - High, RCIC Steam Line Penetration (Drywell Entrance) Area Temperature - High, RCIC Steam Line Torus Room Area Temperature - High, RCIC Equipment Area Temperature - High, RWCU Suction Line Penetration Area Temperature - High, RWCU Heat Exchanger Room Area Temperature - High, RWCU Pump Area Temperature - High (Pumps A and B), and SDC Reactor Pressure - High Functions have been changed based on recent setpoint calculations. Also, three of the above Function titles were changed to be consistent with plant terminology. In addition, the units for the Main Steam Tunnel Radiation - High Functions have been changed to be consistent with the setpoint calculations.	**Specification 3.3.6.1** CTS mark-up p 3 of 25 and 4 of 25 DOCs M14, L16, and L17 (DOCs p 11 of 26, 12 of 26, 13 of 26, 24 of 26, and 25 of 26) NSHCs L16 and L17 (NSHCs p 25 of 32, 26 of 32, 27 of 32, and 28 of 32) ITS mark-up p 3.3-57, 3.3-58, Insert Page 3.3-58, 3.3-59, Insert Page 3.3-59, 3.3-60, Insert Page 3.3-60, 3.3-61, Insert Page 3.3-61 and 3.3-62 JFD DB11 (JFDs p 4 of 5) Retyped ITS p 3.3-52, 3.3-53, 3.3-54, 3.3-55, and 3.3-56

Source of Change	Summary of Change	Affected Pages
New Changes	The acronym for the Reactor Water Cleanup System, "RWC," is being changed to "RWCU." However, it is only being changed in the ITS and ITS Bases mark-ups and the retyped ITS and ITS Bases.	**Specification 3.3.6.1** ITS mark-up p 3.3-61 and Insert p 3.3-61 ITS Bases markup p Insert Page B 3.3-152, B 3.3-155, Insert Page B 3.3-155, B 3.3-169, B 3.3-170, B 3.3-171, B 3.3-172, Insert Page B 3.3-172, and B 3.3-179 Retyped ITS p 3.3-56 Retyped ITS Bases p B 3.3-149, B 3.3-153, B 3.3-168, B 3.3-169, B 3.3-170, B 3.3-171, and B 3.3-179
Typographical Corrections	An incorrect CTS item number was used in a Discussion of Change (Item 9 should have been Item 10). This has been corrected. A incorrect Function reference was used in a Discussion of Change (Function 2.g should have been 2.i). This has been corrected. A typographical error was noted in DOC M14 ("value" has been changed to "values" and the word "voltage" has been deleted). An incorrect DOC number was referenced to a change. This has also been corrected. An ITS Bases mark-up error has been corrected (an Insert page was incorrectly identified).	**Specification 3.3.6.1** CTS mark-up p 23 of 25 DOCs M3, M14, and LA8 (DOCs p 7 of 26, 13 of 26, 15 of 26, and 16 of 26) ITS Bases mark-up p B 3.3-160
RAI 3.3.6.2-1	The ITS did not include the term "automatic" in Condition B, since all secondary containment isolation Functions in the JAFNPP ITS are automatic. The NRC requested a specific JFD be provided for this change. CLB5 has been added to justify the change. The ITS also did not include the term "secondary containment" in ACTION B. The NRC requested that additional clarification be provided for deleting the term or to include the term in the ITS. The term has been added back into the ITS.	**Specification 3.3.6.2** ITS mark-up p 3.3-63 JFD PA1 (deleted) and CLB5 (JFDs p 1 of 3) Retyped ITS p 3.3-58
RAI 3.3.6.2-2 (as modified)	CTS RETs Table 3.10-2 Note (f) provides details of how to perform an LSFT which is not included in the ITS. The deletion is justified as a less restrictive change; however, it is really an administrative change. Therefore, a new A DOC has been provided.	**Specification 3.3.6.2** CTS mark-up p 15 of 15 DOCs A13 and L6 (deleted) (DOCs p 5 of 12 and 11 of 12) NSHC L6 (deleted) (NSHCs p 7 of 9)

Source of Change	Summary of Change	Affected Pages
RAI 3.3.6.1-2	DOC A12 states that the CTS term "trip level settings" is the same as the ITS "Allowable Values". The NRC requested that DOC A12 be clarified to better describe why they are the same. DOC A12 has been modified to provide this requested information. In addition, the Bases have been modified to reflect DOC A12 words.	Specification 3.3.6.2 DOC A12 (DOCs p 4 of 12) ITS Bases mark-up p Insert Page B 3.3-186 Retyped ITS Bases p B 3.3-188
TSTF-205	TSTF-205 has been incorporated into the Bases for SR 3.3.6.2.2, the Channel Functional Test. The TSTF adds an clarification that, in lieu of testing all the required contacts of a channel relay, only a single contact need be tested (i.e., verify change of state of only a single contact).	Specification 3.3.6.2 ITS Bases mark-up p B 3.3-194 and Insert Page B 3.3-194 Bases JFD TA1 (Bases JFDs p 2 of 2) Retyped ITS Bases p B 3.3-195
Amendment 257	This amendment affects the RPS Instrumentation Specification (CTS 3.1.A), which is on a CTS mark-up page used by this Specification. However, the amendment does not affect this Specification; it only results in the renumbering of the CTS mark-up pages.	Specification 3.3.6.2 CTS mark-up p 1 of 15 through 15 of 15
RAI 3.3.6.1-2	DOC A2 states that the CTS term "trip level settings" is the same as the ITS "Allowable Values". The NRC requested that DOC A2 be clarified to better describe why they are the same. DOC A2 has been modified to provide this requested information. In addition, the Bases have been modified to reflect DOC A12 words.	Specification 3.3.7.1 DOC A2 (DOCs p 1 of 5 and 2 of 5) ITS Bases mark-up p Insert Page B 3.3-208 Retyped ITS Bases p B 3.3-199
Amendment 269	This amendment affects the filter train testing requirements of CTS 4.11, which are on CTS mark-up pages used by this Specification. While the amendment does not change any of the requirements of this Specification, it does renumber one of the CTS Surveillances, and also results in renumbering of the CTS mark-up pages.	Specification 3.3.7.1 CTS mark-up p 1 of 8 through 8 of 8 ITS mark-up p 3.3-73
RAI 3.3.6.1-2	DOC A14 states that the CTS term "trip level settings" is the same as the ITS "Allowable Values". The NRC requested that DOC A14 be clarified to better describe why they are the same. DOC A14 has been modified to provide this requested information. In addition, the Bases have been modified to reflect DOC A14 words.	Specification 3.3.7.2 DOC A14 (DOCs p 3 of 8 and 4 of 8) ITS Bases mark-up p Insert Page B 3.3-219c Retyped ITS Bases p B 3.3-204

SUMMARY OF CHANGES TO ITS SECTION 3.3 - REVISION F

Source of Change	Summary of Change	Affected Pages
Typographical correction	The Allowable Value units are changed from "Normal Rated Full Power Background" to "Normal Full Power Background," consistent with the setpoint calculation.	Specification 3.3.7.2 ITS mark-up p Insert Page 3.3-74c Retyped ITS p 3.3-66
New Change	The LCO section of the Bases have been modified, consistent with the changes made for RAI 3.3.1.1-1. In addition, for consistency with other modified DOCs, the Revision of the Engineering Standards Manual has been deleted from DOC M1.	Specification 3.3.7.3 DOC M1 (DOCs p 2 of 5) ITS Bases mark-up p Insert Page B 3.3-219m Retyped ITS Bases p B 3.3-212
Amendment 263	This amendment affects the ECCS Instrumentation Specification (CTS 3.2.B), which is on a CTS mark-up page used by this Specification. However, the amendment does not affect this Specification.	Specification 3.3.8.1 CTS mark-up p 4 of 4
New Change	The Applicable Safety Analyses section of the Bases have been modified, consistent with the changes made for RAI 3.3.1.1-1. In addition, for consistency with other modified DOCs, the Revision of the Engineering Standards Manual has been deleted from DOC L1.	Specification 3.3.8.1 DOC L1 (DOCs p 3 of 3) ITS Bases mark-up p Insert Page B 3.3-221 Retyped ITS Bases p B 3.3-218
RAI 3.3.8.2-1	The CTS does not provide any explicit Applicability requirements for the RPS Electric Power Monitoring Assemblies. The ITS provided an Applicability consistent with the RPS Instrumentation. The NRC requested that additional discussion be provided to justify the proposed Applicability. Additional justification has been provided in DOC L1 and JFD CLB1.	Specification 3.3.8.2 DOC L1 (DOCs p 4 of 7 and 5 of 7) JFD CLB1 (JFDs p 1 of 3 and 2 of 3) Bases JFD CLB2 (Bases JFDs 1 of 2)
RAI 3.3.8.2-2	The CTS requires the plant to be in cold shutdown within 24 hours when the requirements of CTS 3.9.G.1 or 3.9.G.2 are not met. The ITS only requires the plant to be in MODE 3 (hot shutdown). The NRC requested additional justification for this change. Additional justification has been provided in DOC L3.	Specification 3.3.8.2 DOC L3 (DOCs p 6 of 7)

Source of Change	Summary of Change	Affected Pages
RAI 3.3.6.1-4 (should have been 3.3.8.2-4)	DOC A3 states that the CTS term "trip level settings" is the same as the ITS "Allowable Values". The NRC requested that DOC A3 be clarified to better describe why they are the same. DOC A3 has been modified to provide this requested information. In addition, the Bases have been modified to reflect DOC A3 words and, for consistency with other modified DOCs, the Revision of the Engineering Standards Manual has been deleted from DOC M3. Also, the ITS mark-up and retyped ITS are incorrect in that they both show one undervoltage Allowable Value, where the CTS has two Allowable Values, one for Channel A and another for Channel B. The ITS mark-up and Retyped ITS have been corrected.	**Specification 3.3.8.2** DOCs A3 and M3 (DOCs p 1 of 7, 2 of 7, and 3 of 7) ITS mark-up p 3.3-80 ITS Bases mark-up p Insert Page B 3.3-229 Retyped ITS p 3.3-74 Retyped ITS Bases p B 3.3-225
TSTF-205	TSTF-205 has been incorporated into the Bases for SR 3.3.8.2.1, the Channel Functional Test. The TSTF adds an clarification that, in lieu of testing all the required contacts of a channel relay, only a single contact need be tested (i.e., verify change of state of only a single contact).	**Specification 3.3.8.2** ITS Bases mark-up p B 3.3-232 and Insert Page B 3.3-232 Bases JFD TA1 (Bases JFDs p 2 of 2) Retyped ITS Bases p B 3.3-228
Typographical Correction	The term "and MSIV" was inadvertently left in the LCO section of the Bases (it has been removed in all other sections of the Bases). The term has been deleted.	**Specification 3.3.8.2** ITS Bases mark-up p B 3.3-229 Retyped ITS Bases p B 3.3-225

SPLIT REPORT

APPLICATION OF NRC SELECTION CRITERIA TO THE JAMES A. FITZPATRICK NUCLEAR POWER PLANT TECHNICAL SPECIFICATIONS

CONTENTS

1. INTRODUCTION

The purpose of this document is to confirm the results of the BWR Owners' Group (BWROG) application of the Technical Specification selection criteria on a plant specific basis for James A. FitzPatrick Nuclear Power Plant (JAFNPP). New York Power Authority (NYPA) has reviewed the application of the selection criteria to each of the Technical Specifications utilized in BWROG report NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," including Supplement 1 (References 1 and 2, respectively), NUREG-1433, Standard Technical Specifications, General Electric Plants BWR/4," (Reference 3) and applied the criteria to each of the current JAFNPP Technical Specifications (Appendix A and B). Additionally, in accordance with the NRC guidance, this confirmation of the application of selection criteria to JAFNPP includes confirming the risk insights from Probabilistic Risk Assessment (PRA) evaluations, provided in References 1 and 2, as applicable to JAFNPP.

2. SELECTION CRITERIA

New York Power Authority (NYPA) used the selection criteria provided in the NRC Final Policy Statement on Technical Specification Improvements of July 22, 1993 (Reference 4) and 10 CFR 50.36(c)(2)(ii) to develop the results contained in the attached matrix. PRA insights as used in the BWROG submittal were used, confirmed by NYPA, and are discussed in the next section of this report. The selection criteria and discussion provided in the NRC Final Policy statement are as follows:

<u>Criterion 1</u>: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary:

<u>Discussion of Criterion 1</u>: A basic concept in the adequate protection of the public health and safety is the prevention of accidents. Instrumentation is installed to detect significant abnormal degradation of the reactor coolant pressure boundary so as to allow operator actions to either correct the condition or to shut down the plant safely, thus reducing the likelihood of a loss-of-coolant accident.

This criterion is intended to ensure that Technical Specifications control those instruments specifically installed to detect excessive reactor coolant system leakage. This criterion should not, however, be interpreted to include instrumentation to detect precursors to reactor coolant pressure boundary leakage or instrumentation to identify the source of actual leakage (e.g., loose parts monitor, seismic instrumentation, valve position indicators).

<u>Criterion 2</u>: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

<u>Discussion of Criterion 2</u>: Another basic concept in the adequate protection of the public health and safety is that the plant shall be operated within the bounds of the initial conditions assumed in the existing Design Basis Accident and Transient analyses and that the plant will be operated to preclude unanalyzed transients and accidents. These analyses consist of postulated events, analyzed in the FSAR, for which a structure, system, or component must meet specified functional goals. These analyses are contained in Chapters 6 and 14 of the FSAR and are identified as Condition II, III, or IV events (ANSI N18.2) (or equivalent) that either assume the failure of or present a challenge to the integrity of a fission product barrier.

2. (continued)

As used in Criterion 2, process variables are only those parameters for which specific values or ranges of values have been chosen as reference bounds in the Design Basis Accident or Transient analyses and which are monitored and controlled during power operation such that process values remain within the analysis bounds. Process variables captured by Criterion 2 are not, however, limited to only those directly monitored and controlled from the control room. These could also include other features or characteristics that are specifically assumed in Design Basis Accident or Transient analyses even if they cannot be directly observed in the control room (e.g., Linear Heat Generation Rate (LHGR) and Minimum Critical Power Ratio (MCPR)).

The purpose of this criterion is to capture those process variables that have initial values assumed in the Design Basis Accident and Transient analyses, and which are monitored and controlled during power operation. As long as these variables are maintained within the established values, risk to the public safety is presumed to be acceptably low. This criterion also includes active design features (e.g., high pressure/low pressure system valves and interlocks) and operating restrictions (pressure/temperature limits) needed to preclude unanalyzed accidents and transients.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:

Discussion of Criterion 3: A third concept in the adequate protection of the public health and safety is that in the event that a postulated Design Basis Accident or Transient should occur, structures, systems, and components are available to function or to actuate in order to mitigate the consequences of the Design Basis Accident or Transient. Safety sequence analyses or their equivalent have been performed in recent years and provide a method of presenting the plant response to an accident. These can be used to define the primary success paths.

A safety sequence analysis is a systematic examination of the actions required to mitigate the consequences of events considered in the plant's Design Basis Accident and Transient analyses, as presented in Chapters 6 and 14 of the plant FSAR. Such a safety sequence analysis considers all applicable events, whether explicitly or implicitly presented.

The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria), so that the plant response to Design Basis Accidents and Transients limits the consequences of these events to within the appropriate acceptance criteria.

2. (continued)

It is the intent of this criterion to capture into Technical Specifications only those structures, systems, and components that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path for a particular mode of operation does not include backup and diverse equipment (e.g., rod withdrawal block which is a backup to the average power range monitor high flux trip in the startup mode, safety valves which are backup to low temperature overpressure relief valves during cold shutdown).

Criterion 4: A structure, system , or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety:

Discussion of Criterion 4: It is the Commission's policy that licensees retain in their Technical Specifications LCOs, Action statements, and Surveillance Requirements for the following systems (as applicable), which operating experience and PRA have generally shown to be significant to public health and safety and any other structures, systems, or components that meet this criterion:

- Reactor Core Isolation Cooling/Isolation Condenser,
- Residual Heat Removal,
- Standby Liquid Control, and
- Recirculation Pump Trip.

The Commission recognizes that other structures, systems, or components may meet this criterion. Plant-and design-specific PRA's have yielded valuable insight to unique plant vulnerabilities not fully recognized in the safety analysis report Design Basis Accident or Transient analyses. It is the intent of this criterion that those requirements that PRA or operating experience exposes as significant to public health and safety, consistent with the Commission's Safety Goal and Severe Accident Policies, be retained or included in the Technical Specifications.

The Commission expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PRA or risk survey and any available literature on risk insights and PRAs. This material should be employed to strengthen the technical bases for those requirements that remain in Technical Specifications, when applicable, and to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk. Similarly, the NRC staff will also employ risk insights and PRAs in evaluating Technical Specifications related submittals. Further, as a part of the Commissions ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements.

3. PROBABILISTIC RISK ASSESSMENT INSIGHTS

Introduction and Objectives

The Final Policy Statement includes a statement that NRC expects licensees to utilize the available literature on risk insights to verify that none of the requirements to be relocated contain constraints of prime importance in limiting the likelihood or severity of the accident sequences that are commonly found to dominate risk.

Those Technical Specifications proposed for relocation to other plant controlled documents will be maintained under 10 CFR 50.59, safety evaluation review program. These specifications have been compared to a variety of Probabilistic Risk Assessment (PRA) material with two purposes: 1) to identify if a component or variable is addressed by PRA, and 2) to judge if the component or variable is risk-important. In addition, in some cases risk was judged independent of any specific PRA material. The intent of the review was to provide a supplemental screen to the deterministic criteria. Those Technical Specifications proposed to remain part of the Improved Technical Specifications were not reviewed. This review was accomplished in Reference 1 except where discussed in Appendix A, "Justification For Specification Relocation," and has been confirmed by NYPA for those Specifications to be relocated.

Assumptions and Approach

Briefly, the approach used in Reference 1 was the following:

> The risk assessment analysis evaluated the loss of function of the system or component whose LCO was being considered for relocation and qualitatively assessed the associated effect on core damage frequency and offsite releases. The assessment was based on available literature on plant risk insights and PRAs. Table 3-1 lists the PRAs used for making the assessments and is provided at the end of this section. A detailed quantitative calculation of the core damage and offsite release effects was not performed. However, the analysis did provide an indication of the relative significance of those LCOs proposed for relocation on the likelihood or severity of the accident sequences that are commonly found to dominate plant safety risks. The following analysis steps were performed for each LCO proposed for relocation:

3. (continued)

a. List the function(s) affected by removal of the LCO item.

b. Determine the effect of loss of the LCO item on the function(s).

c. Identify compensating provisions, redundancy, and backups related to the loss of the LCO item.

d. Determine the relative frequency (high, medium, and low) of the loss of the function(s) assuming the LCO item is removed from Technical Specifications and controlled by other procedures or programs. Use information from current PRAs and related analyses to establish the relative frequency.

e. Determine the relative significance (high, medium, and low) of the loss of the function(s). Use information from current PRAs and related analyses to establish the relative significance.

f. Apply risk category criteria to establish the potential risk significance or non-significance of the LCO item. Risk categories were defined as follows:

RISK CRITERIA

	Consequence		
Frequency	High	Medium	Low
High	S	S	NS
Medium	S	S	NS
Low	NS	NS	NS

S = Potential Significant Risk Contributor

NS = Risk Non-Significant

g. List any comments or caveats that apply to the above assessment. The output from the above evaluation was a list of LCOs proposed for relocation that could have potential plant safety risk significance if not properly controlled by other procedures or programs. As a result these Specifications will be relocated to other plant controlled documents outside the Technical Specifications.

TABLE 3-1

BWR PRAs USED IN NEDO-31466 (and Supplement 1)
RISK ASSESSMENT

- BWR/6 Standard Plant, GESSAR II, 238 Nuclear Island, BWR/6 Standard Plant Probabilistic Risk Assessment, Docket No. STN 50-447, March 1982.
- La Salle County Station, NEDO-31085, Probabilistic Safety Analysis, February 1988.
- Grand Gulf Nuclear Station, IDCOR, Technical Report 86.2GG, Verification of IPE for Grand Gulf, March 1987.
- Limerick, Docket Nos. 50-352, 50-353, 1981, "Probabilistic Risk Assessment, Limerick Generating Station," Philadelphia Electric Company.
- Shoreham, Probabilistic Risk Assessment Shoreham Nuclear Power Station, Long Island Lighting Company, SAI-372-83-PA-01, June 24, 1983.
- Peach Bottom 2, NUREG-75/0104, "Reactor Safety Study," WASH-1400, October 1975.
- Millstone Point 1, NUREG/CR-3085, "Interim Reliability Evaluation Program: Analysis of the Millstone Point Unit 1 Nuclear Power Plant," January 1983.
- Grand Gulf, NUREG/CR-1659, "Reactor Safety Study Methodology Applications Program: Grand Gulf #1 BWR Power Plant," October 1981.
- NEDC-30936P-A, "BWR Owners' Group Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation) Part 2," June 1987.

4. RESULTS OF APPLICATION OF SELECTION CRITERIA

The selection criteria from Section 2 were applied to the JAFNPP Technical Specifications. The attachment is a summary of that application indicating which Specifications are being retained or relocated. Discussions that document the rationale for the relocation of each Specification which failed to meet the selection criteria are provided in Appendix A. No Significant Hazards Considerations (10 CFR 50.92) evaluations for those Specifications relocated are provided with the Discussion of Changes for the specific Technical Specifications. NYPA will relocate those Specifications identified as not satisfying the criteria to licensee controlled documents whose changes are governed by 10 CFR 50.59.

5. REFERENCES

1. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

2. NEDO-31466, Supplement 1, "Technical Specification Screening Criteria Application and Risk Assessment," February 1990.

3. NUREG-1433, "Standard Technical Specifications, General Electric Plants BWR/4," Revision 1, April 1995.

4. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

ATTACHMENT

SUMMARY DISPOSITION MATRIX

FOR JAFNPP

SUMMARY DISPOSITION MATRIX

Current Number	Title	STS Rev. 4 Number	New TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[a]
1.0	DEFINITIONS	1.0	1.1	Yes	See Note 1, Page 14.
1.1/1.2	SAFETY LIMITS	2.0	2.0		
1.1	Fuel Cladding Integrity				
1.1.A	Reactor Pressure > 785 psig and Core Flow > 10% of Rated	2.1.2	2.1.1.2	Yes	See Note 2, Page 14.
1.1.B	Core Thermal Power Limit, (Reactor Pressure ≤ 785 psig)	2.1.1	2.1.1.1	Yes	See Note 2, Page 14.
1.1.C	Power Transient	None	Deleted	No	Deleted. See Safety Limit technical change discussion in the Discussion of Changes for ITS: Chapter 2.0.
1.1.D	Reactor Water Level (Hot or Cold Shutdown Conditions)	2.1.4	2.1.1.3	Yes	See Note 2, Page 14.
1.2	Reactor Coolant System				
1.2.1	Reactor Vessel Steam Dome Pressure - Irradiated Fuel in Reactor	2.1.3	2.1.2	Yes	See Note 2, Page 14.
1.2.2	Reactor Vessel Steam Dome Pressure - Operating RHR SDC Mode	None	Deleted	No	Deleted. See Safety Limit technical change discussion in the Discussion of Changes for ITS: 3.3.6.1.
2.1/2.2	LIMITING SAFETY SYSTEM SETTINGS				
2.1	Fuel Cladding Integrity				
2.1.A	Trip Settings	2.2.1 3.3.2 3.3.6	3.3.1.1 3.3.6.1	Yes	The application of Technical Specification selection criteria is not appropriate. However, the fuel cladding integrity LSSS have been included as part of the RPS and Primary Containment Isolation Instrumentation Specification, which have been retained since the Functions either actuate to mitigate consequences of Design Basis Accidents (DBAs) and transients or are retained as directed by the NRC as the Functions are part of the RPS. In addition, APRM Rod Block Setting has been relocated (see Appendix A, Page 2).

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

SUMMARY DISPOSITION MATRIX

Current Number	Title	STS Rev. 4 Number	New TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[a]
	LIMITING SAFETY SYSTEM SETTINGS (continued)				
2.2	Reactor Coolant System	2.2.1 3.3.2 3.4.2.1	3.3.1.1 3.3.6.1 3.4.3	Yes	The application of Technical Specification selection criteria is not appropriate. However, the Reactor Coolant System integrity LSSS have been included as part of RPS, Primary Containment Isolation Instrument, and safety relief valve Specifications, which have been retained since the instrument Functions and the safety relief valves mitigate the consequences of DBAs and transients which would result in overpressurization of the RCS or to avoid an inadvertent draindown.
3.0	LIMITING CONDITIONS FOR OPERATION - APPLICABILITY				
3.0.A	Operational Conditions	3.0.1	3.0.1	Yes	See Note 3, Page 14.
3.0.B	Noncompliance	3.0.2	3.0.2	Yes	See Note 3, Page 14.
3.0.C	Generic Actions	3.0.3	3.0.3	Yes	See Note 3, Page 14.
3.0.D	Entry into Operational Conditions	3.0.4	3.0.4	Yes	See Note 3, Page 14.
3.0.E	Power Source Operability Exception	3.8.1	3.8.1	Yes	See Note 3, Page 14. The application of Technical Specification selection criteria is not appropriate. However, this exception to the definition of Operability has been included as part of the Required Actions in LCO 3.8.1.
3.0.F	Equipment Removal from Service	None	3.0.5	Yes	See Note 3, Page 14.
3.0.G	Special Operations	None	3.0.7	Yes	See Note 3, Page 14.
4.0	SURVEILLANCE REQUIREMENTS - APPLICABILITY				
4.0.A	Operational Conditions	4.0.1	SR 3.0.1	Yes	See Note 3, Page 14.
4.0.B	Time of Performance	4.0.2	SR 3.0.2	Yes	See Note 3, Page 14.
4.0.C	Noncompliance	4.0.3	SR 3.0.3	Yes	See Note 3, Page 14.
4.0.D	Entry into Operational Conditions	4.0.4	SR 3.0.4	Yes	See Note 3, Page 14.
4.0.E	Inservice Testing	4.0.5	SR 5.5.7	Yes	Application of the Technical Specification selection criteria is not appropriate. However, Inservice Testing will be included in Technical Specifications as required by 10 CFR 50.36.

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

SUMMARY DISPOSITION MATRIX

Current Number	Title	STS Rev. 4 Number	New TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[a]
3/4.1	REACTOR PROTECTION SYSTEM				
3/4.1.A	Reactor Protection System Instrumentation (Tables 3.1-1, 4.1-1 and 4.1-2)	3/4.3.1	3.3.1.1	Yes-3	Actuates to mitigate consequences of a DBA and/or transients, or it provides an anticipatory scram to ensure the scram discharge volume and thus RPS remains Operable. Functions not specifically credited in the accident analysis are retained for the overall redundancy and diversity of the RPS, as required by licensing basis.
3/4.1.B 4.1.C 4.1.D	Minimum Critical Power Ratio	3/4.2.3 3/4.2.2	3.2.2 3.2.3	Yes-2	Utilized as an initial condition of the design basis transients. Transient analysis are performed to establish the largest reduction in critical power ratio. This value is added to the fuel cladding integrity safety limit to determine the MCPR value.
3/4.2[b]	INSTRUMENTATION	3/4.3	3.3		
3/4.2.A	Primary Containment Isolation Functions (Table 3.2-1, Items 1-20, and Table 4.2-1, Items 1-12)	3/4.3.2	3.3.6.1 3.3.6.2 3.3.7.2	Yes-3,4	Actuates to mitigate the consequences of a DBA LOCA, and a refueling accident, control rod drop accident, or is retained due to risk significance, or is retained as directed by the NRC as it is part of the isolation system.
3/4.2.B	Core and Containment Cooling Systems - Initiation and Control (Table 3.2-2, Items 1-18, and 24 and Table 4.2-2, Items 1-6)	3/4.3.3	3.3.5.1 3.3.5.2 3.3.6.1	Yes-3,4	Actuates to mitigate the consequences of a DBA or small break LOCA, or is being retained due to risk significance.
3/4.2.C	Control Rod Block Actuation (Tables 3.2-3 and 4.2-3)	3/4.3.6	3.3.2.1		
3/4.2.C.1-3	APRMs	3/4.3.6.2	Relocated	No	See Appendix A, Page 1.
3/4.2.C.4-5	Rod Block Monitor	3/4.3.6.1	3.3.2.1.1	Yes-3	Prevents continuous withdrawal of a high worth control rod that could challenge the MCPR Safety Limit and 1 percent cladding plastic strain fuel design limit.
3/4.2.C.6-8	IRMs	3/4.3.6.4	Relocated	No	See Appendix A, Page 2.

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

(b) For current Technical Specification 3/4.2, Instrumentation, when an individual instrument is listed, the current Technical Specification number consists of the Specification number and the instrument's number from the associated 3.2-X Table. For example, the APRM instrument Functions for Control Rod Block Actuation is numbered 3/4.2.C.1-3, where 3/4.2.C is the Specification and "1-3" are the locations of the APRM Instrument Functions in Table 3.2-3 (the APRM instrument Functions include the first through third items in the Table).

SUMMARY DISPOSITION MATRIX

Current Number	Title	STS Rev. 4 Number	New TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[a]
3/4.2[b]	INSTRUMENTATION (continued)				
3/4.2.C.9-10	SRMs	3/4.3.6.3	Relocated	No	See Appendix A, Page 3.
3/4.2.C.11	Scram Discharge Instrument Volume High Water Level	3/4.3.6.5	Relocated	No	See Appendix A, Page 4.
3/4.2.D	Radiation Monitoring Systems - Isolation and Initiation Functions (Appendix B)	NA	NA	NA	The application of Technical Specification selection criteria is evaluated for each Radiological Effluent Technical Specification beginning on Page 11.
3/4.2.E	Drywell Leak Detection (Reactor Coolant Leakage Detection) (Table 3.2-5 and 4.2-5)	3/4.4.3	3.4.6	Yes-1	Leak detection is used to indicate a significant abnormal condition of the reactor coolant pressure boundary.
3/4.2.F	Feedwater Pump Trip and Main Turbine Trip (Tables 3.2-6 and 4.2-6)	3/4.3.9	3.3.2.2	Yes-3	Actuates to limit feedwater addition to the reactor vessel on feedwater controller failure consistent with safety analysis assumptions. Limits neutron flux peak and thermal transient to avoid fuel damage.
3/4.2.G	Recirculation Pump Trip (Tables 3.2-7 and 4.2-7)	3/4.3.4.1	3.3.4.1	Yes-4	RPT is being retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements based on risk significance.
3/4.2.H	Accident Monitoring Instrumentation (Tables 3.2-8 and 4.2-8)	3/4.3.7.5	3.3.3.1	Yes-3	Regulatory Guide 1.97 Type A and Category 1 variables retained. See Appendix A, Page 5 for full discussion of all variables.
3/4.2.I	4kV Emergency Bus Undervoltage Trip (Table 3.2-2 Items 19-23 and Table 4.2-2 Item 7)	3/4.3.3	3.3.8.1	Yes-3	Actuates to mitigate the consequences of design basis accidents during loss of offsite power.
3/4.2.J	Remote Shutdown Capability (Table 3.2.10)	3/4.3.7.4	3.3.3.2	Yes-4	The Remote Shutdown System is considered an important contributor to reducing the risk of accidents; as such, it has been retained in the Technical Specifications as indicated in the NRC Policy Statement.

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

(b) For current Technical Specification 3/4.2, Instrumentation, when an individual instrument is listed, the current Technical Specification number consists of the Specification number and the instrument's number from the associated 3.2-X Table. For example, the APRM instrument Functions for Control Rod Block Actuation is numbered 3/4.2.C.1-3, where 3/4.2.C is the Specification and "1-3" are the locations of the APRM Instrument Functions in Table 3.2-3 (the APRM instrument Functions include the first through third items in the Table).

SUMMARY DISPOSITION MATRIX

Current Number	Title	STS Rev. 4 Number	New TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[a]
3/4.3	REACTIVITY CONTROL	3/4.1	3.1		
3/4.3.A	Reactivity Limitations				
3/4.3.A.1	Reactivity Margin - Core Loading	3/4.1.1	3.1.1	Yes-2	Not a measured process variable, but is important parameter used to confirm the acceptability of the accident analysis.
3/4.3.A.2	Reactivity Margin - Inoperable Control Rods	3/4.1.3.1 3/4.1.3.5 3/4.1.3.7	3.1.3 3.1.5 3.1.8	Yes-3	Control rods are part of the primary success path in mitigating the consequences of DBAs and transients.
3/4.3.B	Control rods				
3/4.3.B.1	Control Rod Coupling	3/4.1.3.6	3.1.3	Yes-3	Control rods are part of the primary success path in mitigating the consequences of DBA's and transients.
3/4.3.B.2	Control Rod Drive Housing Support	3/4.1.3.8	Deleted	No	See CRD Housing Support technical change discussion in the Discussion of Changes for CTS: 3/4.3.B.2.
3/4.3.B.3	Rod Worth Minimizer	3/4.1.4.1	3.3.2.1 3.1.6	Yes-3	Prevents withdrawal of out-of-sequence control rods that might set up high rod worth conditions beyond CRDA assumptions.
3/4.3.B.4	Minimum SRM Count Rate for Rod Withdrawal	3/4.3.7.6	3.3.1.2	Yes	The SRMs have no safety function and are not assumed to function during any DBA or transient analysis. However, the SRMs provide the only on scale monitoring of neutron flux levels during startup and refueling. Therefore, they are being retained in Technical Specifications.
3/4.3.B.5	Operation with a Limiting Control Rod Pattern	3/4.3.6.1	3.3.2.1	Yes-3	Prevents continuous withdrawal of a high worth control rod that would challenge the MCPR Safety Limit and 1 percent cladding plastic strain fuel design limit.
3/4.3.C	Scram Insertion Times	3/4.1.3.2 3/4.1.3.3 3/4.1.3.4	3.1.3 3.1.4 3.1.8	Yes-3	Control rods are part of the primary success path in mitigating the consequences of DBAs and transients.
3/4.3.D	Reactivity Anomalies	3/4.1.2	3.1.2	Yes-2	Confirms assumptions made in the reload safety analysis.
3/4.3.E	Restrictions (Action for 3.3.C and 3.3.D)	3/4.1 (all)	3.1 (all)	Yes-3	The LCOs this Specification is associated with provide the reactivity control requirements that mitigate the consequences of, or prevent a DBA or transient. Therefore, this Specification has been incorporated into ACTIONS for the associated Specifications.

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

SUMMARY DISPOSITION MATRIX

Current Number	Title	STS Rev. 4 Number	New TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[a]
3/4.4	STANDBY LIQUID CONTROL SYSTEM	3/4.1.5	3.1.7	Yes-4	Retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements based on risk significance.
3/4.5	CORE AND CONTAINMENT COOLING SYSTEMS	3/4.5 3/4.7 3/4.8	3.5 3.7 3.8		
3/4.5.A	Core Spray and Low Pressure Coolant Injection (LPCI) Mode of the RHR System	3/4.5.1	3.5.1	Yes-3	Functions to mitigate the consequences of a DBA LOCA.
3/4.5.B	Containment Cooling Mode (of the RHR System)	3/4.7.1.1	3.6.1.9 3.6.2.3 3.7.1	Yes-3	The containment cooling mode of the RHR System is designed for heat removal via RHR heat exchangers following a DBA. As such, acts to mitigate the consequences of an accident.
3/4.5.C	High Pressure Coolant Injection (HPCI System)	3/4.5.1	3.5.1	Yes-3	Functions to mitigate the consequences of small break LOCAs and design basis transients.
3/4.5.D	Automatic Depressurization System (ADS)	3/4.5.1	3.5.1	Yes-3	Functions to mitigate the consequences of small and intermediate LOCAs.
3/4.5.E	Reactor Core Isolation Cooling (RCIC) System	3/4.7.4	3.5.3	Yes-4	Retained in accordance with the NRC Final Policy Statement on Technical Specification Improvements based on risk significance.
3/4.5.F	ECCS - Cold Shutdown	3/4.5.2 3/4.5.3	3.5.2	Yes-3	Functions to mitigate the consequences of a vessel draindown event.
3/4.5.G	Maintenance of Filled Discharge Pipe	3/4.5.1 3/4.5.2 3/4.5.4	3.5.1 3.5.2 3.5.3	Yes-3,4	This Specification ensures the Operability of the ECCS and RCIC, which function to mitigate the consequences of a DBA LOCA, design basis transients (ECCS), or is required to be retained by the NRC Final Policy Statement on Technical Specification Improvements.
3/4.5.H	Average Planar Linear Heat Generation Rate (APLHGR)	3/4.2.1	3.2.1	Yes-2	Peak cladding temperature following a LOCA is primarily dependent on initial APLHGR. As such, it is an initial condition of a DBA analysis.
3.4.5.I	Linear Heat Generation Rate (LHGR)	3/4.2.4	3.2.3	Yes-2	The LHGR limit ensures the fuel design limits are not exceeded anywhere in the core during normal operation including abnormal operation transients.
3/4.5.J	Thermal Hydraulic Stability	None	3.4.1	Yes-2	Assures core conditions are stable, which is assumed in safety analysis. Assure conditions are consistent with those assumed in the safety analysis.

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

SUMMARY DISPOSITION MATRIX

Current Number	Title	STS Rev. 4 Number	New TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[a]
3/4.5	CORE AND CONTAINMENT COOLING SYSTEM (continued)				
3/4.5.K	Single Loop Operation	3/4.4.1	3.4.1	Yes-2	Assure conditions are consistent with those assumed in the safety analysis.
3/4.6	REACTOR COOLANT SYSTEM	3/4.4	3.4		
3.4.6.A	Pressurization and Thermal Limits	3/4.4.6.1	3.4.9	Yes-2	Establishes initial conditions to operation such that operation is prohibited in areas or at temperature rate changes that might cause undetected flaws to propagate in turn challenging the reactor coolant pressure boundary integrity.
3/4.6.B	Deleted in Amendment 158				
3/4.6.C	Specific Activity	3/4.4.5	3.4.6	Yes-2	Specific activity provides an indication of the onset of significant fuel cladding failure and is an initial condition for evaluation of the consequences of an accident due to main steam line break outside containment.
3/4.6.D	Coolant Leakage (Table 4.6-2)	3/4.4.3.1 3/4.4.3.2	3.4.4 3.4.5	Yes-1	Leakage beyond limits would indicate an abnormal condition of the reactor coolant pressure boundary. Operation in this condition may result in reactor coolant pressure boundary failure. Leak detection is used to indicate an abnormal condition of the reactor coolant pressure boundary.
3/4.6.E	Safety/Relief Valves	3/4.4.2.1 3/4.4.2.2	3.4.3	Yes-3	A minimum number of S/RVs is assumed in the safety analysis to mitigate overpressure events.
3/4.6.F	Structural Integrity	3/4.4.8	Relocated	No	See Appendix A, Page 7.
3/4.6.G	Jet Pumps	3/4.4.1.2	3.4.2	Yes-3	Jet pump Operability is assumed in the LOCA analysis to assure adequate core reflood capability.
3/4.6.H	Deleted in Amendment 98				
3/4.6.I	Deleted in Amendment 243				

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

SUMMARY DISPOSITION MATRIX

Current Number	Title	STS Rev. 4 Number	New TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[a]
3/4.7	CONTAINMENT SYSTEMS	3/4.6	3.6		
3/4.7.A	Primary Containment				
3/4.7.A.1	Torus Level and Temperature	3/4.6.2.1	3.6.2.1 3.6.2.2	Yes-2,3	The torus water level and temperature are initial conditions in the DBA LOCA analysis and mitigate the consequences of the DBA.
3/4.7.A.2	Primary Containment Integrity	3/4.6.1.1 3/4.6.1.2 3/4.6.1.3 3/4.6.1.5	3.6.1.1 3.6.1.2 3.6.1.3	Yes-3	Primary containment integrity functions to mitigate the consequences of a DBA. Primary containment leakage is an assumption utilized in the LOCA safety analysis (but is not a process variable). Therefore, it is being retained to ensure primary containment Operability.
3/4.7.A.3	Primary Containment Purge	None	Relocated	No	See Appendix A, Page 8.
3/4.7.A.4	Pressure Suppression Chamber - Reactor Building Vacuum Breakers	3/4.6.4.2	3.6.1.6	Yes-3	Pressure suppression chamber - reactor building vacuum breaker operation is assumed to limit negative pressure differential, secondary to primary containment, that could challenge primary containment integrity.
3/4.7.A.5	Pressure Suppression Chamber - Drywell Vacuum Breakers	3/4.6.4.1	3.6.1.7	Yes-3	Pressure suppression chamber - drywell vacuum breaker operation is assumed in the LOCA analysis to limit drywell to torus differential pressure.
3/4.7.A.6	Oxygen Concentration	3/4.6.6.4	3.6.3.1	Yes-2	Oxygen concentration is limited such that, when combined with hydrogen that is postulated to evolve following a LOCA, the total explosive gas concentration remains below explosive levels. Therefore, primary containment integrity is maintained.
3/4.7.A.7	Drywell - Torus Differential Pressure	None	3.6.2.4	Yes-2	The drywell to torus differential pressure has been established to ensure that appropriate torus and torus support safety margins are maintained following postulated design basis accidents.
3/4.7.A.8	Restrictions (Actions for 3.7.A.1 to 3.7.A.5)	3.6 (all)	3.6 (all)	Yes-2,3	The LCOs associated with this Specification ensure the primary containment capability to mitigate the consequences of a DBA. Therefore, this Specification has been incorporated into Actions for the associated Specification.
3/4.7.B	Standby Gas Treatment System	3/4.6.5.3	3.6.4.3	Yes-3	SGT operation following a DBA acts to mitigate the consequences by treating offsite releases.

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

SUMMARY DISPOSITION MATRIX

Current Number	Title	STS Rev. 4 Number	New TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[a]
3/4.7	CONTAINMENT SYSTEMS (continued)				
3/4.7.C	Secondary Containment	3/4.6.5.1 3/4.6.5.2	3.6.4.1 3.6.4.2	Yes-3	Secondary containment integrity is relied on to limit the offsite dose during an accident by ensuring a release to containment is delayed and treated prior to release to the environment. Valve operation within time limits establishes secondary containment and limits offsite releases to acceptable values.
3/4.7.D	Primary Containment Isolation Valves	3/4.6.3	3.6.1.3	Yes-3	Isolation valves function to limit DBA consequences.
3/4.8	MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES	3/4.7.6	Relocated	No	See Appendix A, Page 9.
3/4.9	AUXILIARY ELECTRICAL SYSTEM	3/4.8	3.8		
3/4.9.A	Normal and Reserve A-C Power Systems	3/4.8.1.1	3.8.1 3.8.7	Yes-3	Functions to mitigate the consequences of a DBA.
3/4.9.B	Emergency A-C Power System	3/4.8.1.1	3.8.1 3.8.3	Yes-3	Functions to mitigate the consequences of a DBA.
3/4.9.C	Diesel Fuel	3/4.8.1.1	3.8.3	Yes-3	Required to ensure diesel fuel to the emergency diesel generators. As such, functions to mitigate the consequences of a DBA.
3/4.9.D	Diesel Generator Operability (Shutdown)	3/4.8.1.2	3.8.2	Yes-3	Functions to mitigate the consequences of a vessel draindown event and is needed to support NRC Final Policy Statement requirement for decay heat removal.
3/4.9.E	Station Batteries	3/4.8.2.1	3.8.4 3.8.6 3.8.7	Yes-3	Functions to mitigate the consequences of a DBA.
3/4.9.F	LPCI MOV Independent Power Supplies	None	3.5.1 3.8.4 3.8.6	Yes-3	Functions to mitigate the consequences of a DBA.
3/4.9.G	Reactor Protection System Electrical Protection Assemblies	3/4.8.4.4	3.3.8.2	Yes-3	Provides protection for the RPS bus powered components against unacceptable voltage and frequency conditions that could degrade the components so that it would not perform the intended safety function.

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

SUMMARY DISPOSITION MATRIX

Current Number	Title	STS Rev. 4 Number	New TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[a]
3/4.10	CORE ALTERATIONS	3/4.9	3.9		
3/4.10.A.1	Refueling Interlocks	Table 1.2 3/4.9.1	3.9.1 3.9.2	Yes-3	Provides an interlock to preclude fuel loading with a control rod withdrawn. Operation is assumed in the event of a control rod removal error during refueling and fuel assembly insertion error in the refueling accident analysis.
3/4.10.A.2	Fuel Loading	3/4.9.3	3.9.3 3.10.6	Yes-2	All control rods are required to be fully inserted when loading fuel. This requirement is assumed as an initial condition in the event of a fuel assembly insertion error in the refueling accident analysis.
3/4.10.A.3,4	Refueling Equipment Hoist Loaded	3/4.9.1	Deleted	No	Deleted. See Refuel Equipment Interlocks technical change discussion in the Discussion of Changes for ITS 3.9.1.
3/4.10.A.5	Multiple Control Rod/Control Rod Drive - Withdrawal or Removal	3/4.9.10.2	3.10.6	Yes	See Note 4, Page 14.
3/4.10.A.6,7	Spiral off-load/on-load	3/4.9.10.2	3.10.6	Yes	See Note 4, Page 14.
3/4.10.B	Core Monitoring	3/4.9.2	3.3.1.2	Yes	The SRMs have no safety function and are not assumed to function during any DBA or transient analysis. However, the SRMs provide the only on scale monitoring of neutron flux levels during startup and refueling. Therefore, they are being retained in Technical Specifications.
3/4.10.C	Spent Fuel Storage Pool Water Level	3/4.9.9	3.7.7	Yes-2	A minimum amount of water is required to assure adequate scrubbing of fission products following a refueling accident.
3/4.10.D	Control Rod and Control Rod Drive Maintenance	3/4.9.10.1	3.10.5	Yes	See Note 4, Page 14.
3/4.11	ADDITIONAL SAFETY RELATED PLANT CAPABILITIES				
3/4.11.A	Main Control Room Ventilation	3/4.7.2	3.3.7.1 3.7.3	Yes-3	Maintains habitability of the control room so that operators can remain in the control room following an accident. As such, it mitigates the consequences of an accident by allowing operators to continue accident mitigation activates from the control room.
3/4.11.B	Deleted in Amendment 231				

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

SUMMARY DISPOSITION MATRIX

Current Number	Title	STS Rev. 4 Number	New TS Number	Retained/ Criterion for Inclusion	Bases for Inclusion/Exclusion[a]
3/4.11	ADDITIONAL SAFETY RELATED PLANT CAPABILITIES (continued)				
3/4.11.C	Battery Room Ventilation	None	Deleted	No	Deleted. See Battery Room Ventilation technical change discussion in the Discussion of Changes for CTS 3/4.11.C.
3/4.11.D	Emergency Service Water System	3/4.7.1.2	3.3.7.3 3.7.2	Yes-3	Designed to supply lake water to safe shutdown loads following an accident. As such acts to mitigate the consequences of an accident.
3/4.11.E	Intake Deicing Heater	None	3.7.2	Yes-3	Ensures an adequate supply of lake water is available to support safe shutdown loads.
3/4.12	Special Operation				
3/4.12.A	Inservice Leak and Hydrostatic Testing Operation	None	3.10.1	Yes	Although this Specification does not meet any criteria of the NRC Final Policy Statement, it has been retained since it provides flexibility to perform certain operations by appropriately modifying requirements of other LCOs.
5.0	Design Features	5.0	4.0	Yes	Application of Technical Specification selection criteria is not appropriate. However, Design Features will be included in Technical Specifications as required by 10 CFR 50.36.
6.0	Administrative Controls	6.0	5.0	Yes	Application of Technical Specification selection criteria is not appropriate. However, Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.
7.0	References	None	Deleted	No	Application of Technical Specification selection criteria is not appropriate. The appropriate references will be incorporated in the expanded improved Bases.

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

SUMMARY DISPOSITION MATRIX

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	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.
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.

SUMMARY DISPOSITION MATRIX

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 (continued)				
3.7	Offgas Treatment System Explosive Gas Mixture Instrumentation	None	5.5.9	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.
3.8	Standby Gas Treatment System (SGTS) (Table 3.10-1 and 3.10-2)	3/4.2	3.3.6.2	Yes-3	Actuates to mitigate the consequences of a DBA LOCA or a refueling accident.
3.9	Mechanical Vacuum Pump Isolation (Table 3.10-1 and 3.10-2)	None	3.3.7.2	Yes-3	Assumed to function to mitigate the consequences of a refueling accident.
3.10	Main Control Room Ventilation Radiation Monitor (Table 3.10-1 and 3.10-2)	None	3.3.7.1		Alarms during design basis events so that operators can place Control Room Emergency Ventilation Air Supply System in isolate mode to ensure control room dosage remains within limits.
4.0	Solid Radioactive Waste				
4.1	Process Control Program	None	Relocated	No	See Appendix A, Page 19.
5.0	Total Dose				
5.1	Total Dose from Uranium Fuel Cycle	None	Relocated	No	See Appendix A, Page 20.
6.0	Radiological Environmental Monitoring				
6.1	Monitoring Program	None	Relocated	No	See Appendix A, Page 21.
6.2	Land Use Census Program	None	Relocated	No	See Appendix A, Page 22.
6.3	Interlaboratory Comparison Program	None	Relocated	No	See Appendix A, Page 23.
7.0	Administrative Controls	6.0	5.0	Yes	Application of Technical Specification selection criteria is not appropriate. However, Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.

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

SUMMARY DISPOSITION MATRIX

NOTE 1: DEFINITIONS

This section provides definitions for several defined terms used throughout the remainder of Technical Specifications. They are provided to improve the meaning of certain terms. As such, direct application of the Technical Specification selection criteria is not appropriate. However, only those definitions for defined terms that remain as a result of application of the selection criteria, will remain as definitions in this section of Technical Specifications. In addition, this section provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operation and Surveillance Requirements. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements will be added to the JAFNPP Technical Specifications consistent with NUREG-1433, Revision 1.

NOTE 2: SAFETY LIMITS/LSSS

Application of Technical Specification selection criteria is not appropriate. However, Safety Limits and Limiting Safety System Settings (as part of Reactor Protection System, ECCS, and Primary Containment Isolation Instrumentation) will be included in Technical Specifications as required by 10 CFR 50.36.

NOTE 3: 3.0/4.0

These Specifications provide generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operation and Surveillance Requirements. As such, direct application of the Technical Specification selection criteria is not appropriate. However, the general requirements of 3.0/4.0 will be retained in Technical Specifications, as modified consistent with NUREG-1433, Revision 1.

NOTE 4: SPECIAL TEST EXCEPTIONS

These Specifications are provided to allow relaxation of certain Limiting Conditions for Operation under certain specific conditions to allow testing and maintenance. They are directly related to one or more Limiting Conditions for Operations. Direct application of the Technical Specification selection criteria is not appropriate. However, those special test exceptions, directly tied to Limiting Conditions for Operation that remain in Technical Specifications, will also remain as Technical Specifications.

APPENDIX A

JUSTIFICATION FOR

SPECIFICATION RELOCATION

3/4.2.C CONTROL ROD BLOCK ACTUATION

LCO Statement:

The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2-3.

3/4.2.C.1 APRM Flow Referenced Neutron Flux
3/4.2.C.2 APRM Neutron Flux-Start-up
3/4.2.C.3 APRM Downscale

Discussion:

The Average Power Range Monitor (APRM) control rod blocks function to limit control rod withdrawal errors during power range operations utilizing LPRM signals to create the APRM rod block signal. APRMs provide information about the average core power and APRM rod blocks are not used to mitigate a design basis accident (DBA) or transient.

Comparison to Screening Criteria:

1. The APRM Control Rod Block instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The APRM Control Rod Block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analysis.

3. The APRM Control Rod Block instrumentation is not a part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 135) of NEDO-31466, the loss of the APRM Control Rod Block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Control Rod Block LCO and Surveillances applicable to APRM Instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2.C CONTROL ROD BLOCK ACTUATION

LCO Statement:

The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2-3.

3/4.2.C.6 IRM Detector not in Start-up Position
3/4.2.C.7 IRM Upscale
3/4.2.C.8 IRM Downscale

Discussion:

The Intermediate Range Monitor (IRM) control rod blocks function to limit control rod withdrawal errors during reactor startup utilizing IRM signals to create the rod block signal. IRMs are provided to monitor the neutron flux levels during refueling, shutdown, and startup conditions. No design basis accident (DBA) or transient analysis takes credit for rod block signals initiated by IRMs.

Comparison to Screening Criteria:

1. The IRM Control Rod Block instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The IRM Control Rod Block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analysis.

3. The IRM Control Rod Block instrumentation is not a part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 138) of NEDO-31466, the loss of the IRM Control Rod Block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Control Rod Block LCO and Surveillances applicable to IRM Instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2.C CONTROL ROD BLOCK ACTUATION

LCO Statement:

The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2-3.

3/4.2.C.9 SRM Detector not in Start-up Position
3/4.2.C.10 SRM Upscale

Discussion:

The Source Range Monitor (SRM) control rod blocks function to limit control rod withdrawal errors during reactor startup utilizing SRM signals to create the rod block signal. SRM signals are used to monitor neutron flux during refueling, shutdown and startup conditions. No design basis accident (DBA) or transient analysis takes credit for rod block signals initiated by the SRMs.

Comparison to Screening Criteria:

1. The SRM Control Rod Block instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The SRM Control Rod Block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analysis.

3. The SRM Control Rod Block instrumentation is not a part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 137) of NEDO-31466, the loss of the SRM Control Rod Block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Control Rod Block LCO and Surveillances applicable to SRM Instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2.C CONTROL ROD BLOCK ACTUATION

LCO Statement:

The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2-3.

3/4.2.C.11 Scram Discharge Instrument Volume High Water Level

Discussion:

The Scram Discharge Instrument Volume High Water Level (SDVHWL) control rod block functions to prevent control rod withdrawals, utilizing SDVHWL signals to create the rod block signal if water is accumulating in the scram discharge instrument volume. The purpose of measuring the scram discharge instrument volume water level is to ensure that there is sufficient volume to contain the water discharged by the control rod drives during a scram, thus ensuring that the control rods will be able to insert fully. This rod block signal provides an indication to the operator that water is accumulating in the scram discharge instrument volume and prevents further rod withdrawals. With continued water accumulation, a reactor protection system initiated scram signal will occur. Thus, the SDVHWL rod block signal provides an opportunity for the operator to take action to avoid a subsequent scram. No design basis accident (DBA) or transient takes credit for rod block signals initiated by the SDVHWL instrumentation.

Comparison to Screening Criteria:

1. The SDVHWL Control Rod Block instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The SDVHWL Control Rod Block instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analysis.

3. The SDVHWL Control Rod Block instrumentation is not a part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 139) of NEDO-31466, the loss of the SDVHWL Control Rod Block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Control Rod Block LCO and Surveillances applicable to SDVHWL Instrumentation may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.2.H ACCIDENT MONITORING INSTRUMENTATION

LCO Statement:

The limiting conditions for operation for the instrumentation that provides accident monitoring are given in Table 3.2-8.

Discussion:

Each individual accident monitoring parameter has a specific purpose, however, the general purpose for all accident monitoring instrumentation is to provide sufficient information to confirm an accident is proceeding per prediction, i.e. automatic safety systems are performing properly, and deviations from expected accident course are minimal.

Comparison to Screening Criteria:

The NRC position on application of the screening criteria to post-accident monitoring instrumentation is documented in letter dated May 7, 1988 from T.E. Murley (NRC) to R.F. Janecek (BWROG). The position was that the post-accident monitoring instrumentation table list should contain, on a plant specific basis, all Regulatory Guide 1.97 Type A instruments specified in the plant's Safety Evaluation Report (SER) on Regulatory Guide 1.97, and all Regulatory Guide 1.97 Category 1 instruments. Accordingly, this position has been applied to the JAFNPP Regulatory Guide 1.97 instruments. Those instruments meeting these criteria have been retained in Technical Specifications. The instruments not meeting these criteria will be relocated from the Technical Specifications to plant controlled documents.

The following summarizes the JAFNPP position for those instruments currently in Technical Specifications.

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

Type A Variables

- Containment High Range Radiation Monitor
- Drywell Pressure (narrow range)
- Drywell Pressure (wide range)
- Drywell Temperature
- Torus Water Level (wide range)
- Torus Bulk Water Temperature
- Torus Pressure
- Primary Containment Hydrogen/Oxygen Concentration
- Reactor Vessel Pressure
- Reactor Water Level (fuel zone)
- Reactor Water Level (wide range)
- Core Spray Flow*
- Core Spray Discharge Pressure*
- LPCI (RHR) Flow*
- RHR Service Water Flow*

RAI 3.3.3.1-9

*As part of this ITS conversion, these 4 variables are reclassified from Type A and Category 1 to Type D and Category 2. The justification for this reclassification is provided in ITS 3.3.3.1, DOC R2.

3/4.2.H ACCIDENT MONITORING INSTRUMENTATION (continued)

Other Type, Category 1 Variables

There are no additional instruments that fall under this category.

For other post-accident monitoring instrumentation currently in Technical Specifications, their loss is not considered risk significant since the variable they monitor does not qualify as a Type A or Category 1 variable (one that is important to safety, and needed by the operator so that the operator can perform necessary manual actions).

Conclusion:

Since the screening criteria have not been satisfied for non-Regulatory Guide 1.97 Type A or Category 1 variable instruments, their associated LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications. The instruments to be relocated are as follows:

- Stack High Range Effluent Monitor
- Turbine Building Vent High Range Effluent Monitor
- Radwaste Building Vent High Range Effluent Monitor
- Safety/Relief Valve Position Indicator
- Torus Water Level (narrow range)
- Drywell – Torus Differential Pressure
- Core Spray Flow
- Core Spray Discharge Pressure
- LPCI (RHR) Flow
- RHR Service Water Flow

3/4.6.F STRUCTURAL INTEGRITY

LCO Statement:

The structural integrity of the Reactor Coolant System shall be maintained at the level required by the original acceptance standards throughout the life of the plant.

Discussion:

The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained throughout the components life. Other Technical Specifications require important systems to be Operable (for example, ECCS 3/4.5.A) and in a ready state for mitigative action. This Technical Specification is more directed toward prevention of component degradation and continued long term maintenance of acceptable structural conditions. Hence, it is not necessary to retain this Specification to ensure immediate Operability of safety systems.

Further, this Technical Specification prescribes inspection requirements which are performed during plant shutdown. It is therefore not directly important for responding to DBAs.

Comparison to Screening Criteria:

1. The inspections stipulated by this Specification are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The inspections stipulated by this Specification do not monitor process variables that are initial assumptions in a DBA or transient analysis.

3. The ASME Code Class 1, 2, and 3 components inspected per this Specification are assumed to function to mitigate a DBA. Their capability to perform this function is addressed by other Technical Specifications. This Technical Specification, however, only specifies inspection requirements for these components; and these inspections can only be performed when the plant is shutdown. Therefore, Criterion 3 is not satisfied.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 216) of NEDO-31466, the assurance of operability of the entire system as verified in the system Operability Specification dominates the risk contribution of the system. As such, the lack of a long term assurance of Inservice Inspection Specification was found to be a non-significant risk contributor to core damage frequency and offsite releases. Furthermore, the requirement is currently covered by 10 CFR 50.55a and the plant's Inservice Inspection Program. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Structural Integrity LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.7.A.3 PRIMARY CONTAINMENT PURGE

LCO Statement:

The containment shall be purged through the Standby Gas Treatment System whenever the primary containment integrity is required. If this requirement cannot be met, then purging shall be discontinued without delay.

Discussion:

The drywell vent and purge system is used primarily to control drywell-to-suppression chamber differential pressure during reactor operation, to reduce drywell airborne radioactivity levels before personnel entry and to purge the nitrogen from the drywell for personnel safety. This LCO is intended to provide reasonable assurance that releases from normal drywell purging operations will not exceed the annual dose limits of 10 CFR Part 20 for unrestricted areas. These limits are not related to protection of the public from the consequences of any DBA or transient. The acceptability of the relocation of this Specification from the plant Technical Specifications has already been endorsed by the NRC as indicated in Generic Letter 89-01.

Comparison to Screening Criteria:

1. Purging of the primary containment through the Standby Gas Treatment System is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. Purging of the containment through the Standby Gas Treatment System has no relationship to any process variable that is an initial condition of a DBA or transient analysis.

3. Purging through the Standby Gas Treatment System during normal operation is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 6.0 and summarized in Table 6-1 (Item 318) of NEDO-31466, Supplement 1, venting or purging of the drywell, as controlled by this specification, was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Primary Containment Purge LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3/4.8 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

LCO Statement:

Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material, shall have removable contamination of less than or equal to 0.005 microcuries.

Discussion:

The limitations on miscellaneous radioactive materials sources are intended to ensure that the total body or individual organ irradiation doses does not exceed allowable limits in the event of ingestion or inhalation. This is done by imposing a maximum limitation of $\leq$ 0.005 microcuries of removable contamination on each sealed source. This requirement and the associated Surveillance Requirements bear no relation to the conditions or limitations which are necessary to ensure safe reactor operation.

Comparison to Screening Criteria:

1. Miscellaneous radioactive materials sources are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. Miscellaneous radioactive materials sources are not a process variable that is an initial condition of a DBA or transient analysis.

3. Miscellaneous radioactive materials sources are not used in any part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 267) of NEDO-31466, the Miscellaneous radioactive materials sources being not within limits was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Miscellaneous Radioactive Materials Sources LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

2.1 LIQUID EFFLUENT MONITORS

LCO Statement:

The limiting conditions for operation of the instruments that monitor radioactive liquid effluent are given in Table 2.1-1. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the Offsite Dose Calculation Manual (ODCM) without delay suspend the release of radioactive liquid effluent monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

Discussion:

The radioactive liquid effluent monitoring instrumentation is neither a safety system nor is it connected to the reactor coolant system. This instrumentation is used for the purpose of showing conformance to the discharge limits of 10 CFR Part 20. It is not installed to detect excessive reactor coolant leakage. The radioactive liquid effluent monitors are used routinely to provide continuous check on the release of radioactive liquid effluent from the normal plant liquid effluent flowpaths. These Technical Specifications require the Licensee to maintain operability of various liquid effluent monitors and establish setpoints in accordance with the ODCM. The alarm/trip setpoints are established to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. Plant design basis accident (DBA) analyses do not assume any action, either automatic or manual, resulting from radioactive liquid effluent monitors.

Comparison to Screening Criteria:

1. The radioactive liquid effluent monitoring instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The radioactive liquid effluent monitoring instrumentation is not used to monitor a process variable that is an initial condition of a DBA or transient analysis.

3. The radioactive liquid effluent monitoring instrumentation is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 3.5 and 6 and summarized in Table 4-1 (Item 188) of NEDO 31466, the loss of radioactive liquid effluent monitoring instrumentation was found to be a non-significant risk contributor to core damage and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with this assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Radioactive Liquid Effluent Monitoring Instrumentation LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

2.2 CONCENTRATION OF LIQUID EFFLUENTS

LCO Statement:

The concentration of radioactive materials released to the unrestricted areas shall not exceed the values specified in 10 CFR Part 20, Appendix B, Table II, Column 2. For dissolved or entrained noble gases, the concentration shall be limited to 2 x 10^{-4} microcurie/ml.

Discussion:

10 CFR Part 20, BII(2) refers to releases to an unrestricted area of radioactive material in concentrations that exceed the specified limits. No screening criteria apply because the process variable of the LCO is not an initial condition of a design basis accident (DBA) or transient analysis. Neither does the system comprise a part of the safety sequence analysis or a part of the primary coolant pressure boundary. Effluent control is for protection against radiation hazards from licensed activities, not accidents.

Comparison to Screening Criteria:

1. The concentration of liquid effluents limits are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The concentration of liquid effluents limits are not a process variable that is an initial condition of a DBA or transient analysis.

3. The concentration of liquid effluents limits are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 296) of NEDO-31466, liquid releases during normal operation are a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Concentration LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

2.3 DOSE FROM LIQUID EFFLUENTS

LCO Statement:

The dose to a member of the public from radioactive materials released from the plant in liquid effluents to unrestricted areas shall be limited as follows:

a. During any calendar quarter, limited to less than or equal to 1.5 mrem to the whole body and to less than or equal to 5 mrem to any organ; and,

b. During any calendar year, limited to less than or equal to 3 mrem to the whole body and to less than or equal to 10 mrem to any organ.

Discussion:

Limitations of the quarterly and annual projected doses to members of the public which results from cumulative liquid effluent discharges during normal operation over extended periods is intended to assure compliance with the dose objectives of 10 CFR Part 50, Appendix I. These limits are not related to protection of the public from any design bases accident or transient.

Comparison to Screening Criteria:

1. The dose limits from liquid effluents are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant boundary prior to a DBA.

2. The dose limits from liquid effluents are not a process variable that is an initial condition of a DBA or transient analysis.

3. The dose limits from liquid effluents are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 297) of NEDO-31466, dose limits from liquid effluents were found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Dose from Liquid Effluents LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

2.4 LIQUID RADIOACTIVE WASTE TREATMENT SYSTEM

LCO Statement:

The liquid radioactive waste treatment system shall be used when the projected dose from untreated liquid releases, over a 31 day period, to a member of the public would exceed:

1. 0.06 mrem to the whole body; or,
2. 0.2 mrem to any organ.

Discussion:

The requirement for a liquid waste treatment system in 10 CFR Part 50, Appendix A, GDC 60, pertains to controlling the release of site liquid effluents during normal operational occurrences. No loss of primary coolant is involved; neither is an accident condition assumed or implied. The limits for release in 10 CFR Part 50, Appendix I, Sec. II.A, for liquids are design objectives for operation.

Comparison to Screening Criteria:

1. The Liquid Radioactive Waste Treatment System is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The Liquid Radioactive Waste Treatment System is not used to monitor a process variable that is an initial condition of a DBA or transient analysis.

3. The Liquid Radioactive Waste Treatment System is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 298) of NEDO-31466, the loss of the liquid radioactive waste treatment system was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this revaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Liquid Radioactive Waste Treatment System LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3.1 GASEOUS EFFLUENT MONITORS

LCO Statement:

Radioactive gaseous waste released to the environment via the below listed pathways shall be monitored and recorded during release from the respective pathway.

1. Main stack exhaust
2. Refuel floor exhaust
3. Reactor building exhaust
4. Turbine building exhaust
5. Radwaste building exhaust

Discussion:

The radioactive gaseous effluent monitors are neither a safety system nor is it 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 radioactive gaseous effluent monitors are used routinely to provide continuous check on the releases of radioactive gaseous effluents from the normal plant gaseous effluent flow paths. These Technical Specifications require the Licensee to maintain operability of various effluent monitors and establish setpoints in accordance with the ODCM. The alarm/trip setpoints 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 radioactive effluent monitors (except as indicated in the Discussion of Changes for ITS: 3.3.6.2, Secondary Containment Instrumentation. The Refuel Floor, and Reactor Building exhaust monitor are retained in ITS: 3.3.6.2.

Comparison to Screening Criteria:

1. The Gaseous Effluent 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 Gaseous Effluent Monitors do not monitor a process variable that is an initial condition of a DBA or transient.

3. The Gaseous Effluent 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 Radioactive Gaseous Effluent Monitor was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to the pathways associated with the main stack, turbine building, and radwaste building exhaust, and concurs with the assessment.

Conclusion:

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

3.2 GASEOUS DOSE RATE

LCO Statement:

The dose rate at or beyond the site boundary due to radioactive materials released from the plant in gaseous effluents shall be limited as follows:

1. $\leq$ 500 mrem/year to the whole body and $\leq$ 3000 mrem/year to the skin from noble gases; and,

2. $\leq$ 1500 mrem/year to any organ from Iodine-131, Iodine-133, Tritium and for radioactive materials in particulate form with half-lives greater than 8 days (inhalation pathway only).

Discussion:

This LCO limits the dose rate due to gaseous effluents in unrestricted areas at any time to a value less than the yearly dose limit of 10 CFR Part 20. This provides reasonable assurance that no member of the public is exposed to annual average concentrations which exceed the limits of 10 CFR Part 20 Appendix B, Table-II. This is a limit which applies to normal operation of the plant. It is not assumed as an initial condition of any design basis accident or transient and is not relied upon to limit the consequences of such events.

Comparison to Screening Criteria:

1. Gaseous dose rate limits are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The gaseous dose rate limits are not a process variable that is an initial condition of a DBA or transient analysis.

3. The gaseous dose rate limits are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 300) of NEDO-31466, the gaseous dose rate limit was found to be a non-significant risk contributor to core damage frequency and offsite releases during operation. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Gaseous Dose Rate LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3.3 AIR DOSE, NOBLE GASES

LCO Statement:

The air dose to areas at or beyond the site boundary from noble gases released from the plant in gaseous effluents shall be limited:

a. During any calendar quarter, to less than or equal to 5 mrad for gamma radiation, and less than or equal to 10 mrad from beta radiation; and

b. During any calendar year, to less than or equal to 10 mrad from gamma radiation and less than or equal to 20 mrad from beta radiation.

Discussion:

Limitation of the quarterly and annual air doses from noble gases in plant gaseous effluents during normal operation over extended periods is intended to assure compliance with the dose objectives of 10 CFR Part 50, Appendix I. These limits are not related to protection of the public from the consequences of any design basis accident or transient.

Comparison to Screening Criteria:

1. The air dose noble gas limits are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The air dose noble gas limits are not a process variable that is an initial condition of a DBA or transient analysis.

3. The air dose noble gas limits are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (item 301) of NEDO-31466, the air dose noble gas limits were found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Air Dose, Noble Gases LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3.4 DOSE DUE TO IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

LCO Statement:

The dose to a member of the public at or beyond the site boundary from Iodine-131, Iodine-133, Tritium, and radionuclides in particulate form with half-lives greater than 8 days released from the plant in gaseous effluents shall be limited:

1. During any calendar quarter to less than or equal to 7.5 mrem to any organ; and,

2. During any calendar year to less than or equal to 15 mrem to any organ.

3. Less than 0.1% of the limits of Specification 3.4.a.1 and 3.4.a.2 as a result of burning contaminated oil.

Discussion:

Limitation of the quarterly and annual projected doses to members of the public from radionuclides other than noble gases during normal operation over extended periods is intended to assure compliance with the dose objectives of 10 CFR Part 50, Appendix I. These limits are not related to protection of the public from the consequences of any design basis accident or transient.

Comparison to Screening Criteria:

1. The dose due to iodine-131, iodine-133, tritium, and radioactive material in particulate form limits are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The dose due to iodine-131, iodine-133, tritium, and radioactive material in particulate form limits are not a process variable that is an initial condition of a DBA or transient analysis.

3. The dose due to iodine-131, iodine-133, tritium, and radioactive material in particulate form limits are not utilized in any capacity in a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 302) of NEDO-31466, the dose due to iodine-131, iodine-133, tritium, and radionuclides in particulate form releases during normal operations were found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Dose due to Iodine-131, Iodine-133, Tritium, and Radioactive Material In Particulate Form LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

3.6 OFFGAS TREATMENT SYSTEM

LCO Statement:

The Offgas Treatment System shall be used to reduce the concentration of radioactive materials in gaseous effluents prior to release from the plant within 24 hours after the start-up of the second turbine driven feedwater pump.

Discussion:

The Offgas Treatment System reduces the activity level of the non-condensible fission product gases from fuel defects removed from the main condenser prior to their release to the environs. The Operability of the Offgas Treatment System is required to meet the requirements of 10 CFR 50.36a and General Design Criteria 60 of Appendix A to 10 CFR Part 50 (i.e., releases of radioactive materials in gaseous effluents will be kept "as low as reasonably achievable"). The Operability of the Offgas Treatment System is not assumed in the analysis of any design bases accident or transient. However, offgas activity is an initial condition of a design basis accident and is being retained in ITS LCO 3.7.5. Therefore, there is no need to retain this requirement.

Comparison to Screening Criteria:

1. The Offgas Treatment System is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. Although offgas activity is an initial condition of a DBA and does satisfy Criterion 2, this process variable is addressed by another Technical Specification. The Offgas Treatment System is not used to monitor any other process variable that is an initial condition of a DBA or transient analysis. As such, Criterion 2 is not satisfied.

3. The Offgas Treatment System is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 303) of NEDO-31466, the loss of the Offgas Treatment System was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Offgas Treatment System LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

4.1 SOLID RADIOACTIVE WASTE - PROCESS CONTROL PROGRAM

LCO Statement:

The solid radwaste system shall be used in accordance with the PCP to process wet radioactive wastes to meet shipping and burial ground requirements.

Discussion:

The Solid Radwaste System is a logical continuation of the liquid radwaste system. It operates on the same requirement for effluent control, identified as 10 CFR Part 50, Appendix A, GDC 60. The system serves to control operational release of solid waste, not accidental release.

Comparison to Screening Criteria:

1. The Solid Radwaste System is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The Solid Radwaste System does not monitor a process variable that is an initial condition of a DBA or transient analysis.

3. The Solid Radwaste System is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 308) of NEDO-31466, the Solid Radwaste System was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Solid Radioactive Waste-Process Control Program LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

5.1 TOTAL DOSE - TOTAL DOSE FROM URANIUM FUEL CYCLE

LCO Statement:

The dose or dose commitment to any member of the public, due to releases of radioactivity and radiation, from uranium fuel cycle sources shall be limited as follows:

1. Less than or equal to 25 mrem/year to the whole body; and,
2. Less than or equal to 25 mrem/year to any organ except the thyroid which shall be limited to less than or equal to 75 mrem/year.

Discussion:

This LCO limits the annual doses to individual members of the public from all plant sources. This is intended to assure that normal operation of the plant is in compliance with the provisions of 40 CFR Part 190. These limits are not related to protection of the public from any design basis accident or transient.

Comparison to Screening Criteria:

1. The total dose from uranium fuel cycle are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. The total dose from uranium fuel cycle are not a process variable that is an initial condition of a DBA or transient analysis.
3. The total dose from uranium fuel cycle are not part of a primary success path in the mitigation of a DBA or transient.
4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 304) of NEDO-31466, the effluent dose liquid/gaseous limits were found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Total Dose from the Uranium Fuel Cycle LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

6.1 RADIOLOGICAL ENVIRONMENTAL MONITORING - MONITORING PROGRAM

LCO Statement:

With the radiological environmental monitoring program not being conducted as specified in Table 6.1-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

Discussion:

The radiological environmental monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public resulting from station operations. This program monitors the long term impact of normal plant operations.

Comparison to Screening Criteria:

1. The radiological environmental monitoring program is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The radiological environmental monitoring program does not monitor a process variable that is an initial condition of a DBA or transient analysis.

3. The radiological environmental monitoring program is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 309) of NEDO-31466, not conducting a radiological environmental monitoring program was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Monitoring Program LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

6.2 LAND USE CENSUS PROGRAM

LCO Statement:

A land use census shall be conducted and shall identify the locations of all milch animals, the nearest residence, and all gardens of greater than 50 square meters producing fresh leafy vegetables, in each of the 16 meteorological sectors within a distance of 5 miles from the site.

Discussion:

The land use census required by this specification supports the measurement of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public resulting from station operations. This program ensures that changes in the use of areas at or beyond the site boundary are identified and changes made to the radiological environmental monitoring program, if required.

Comparison to Screening Criteria:

1. The land use census is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The land use census is not a process variables that is an initial condition of a DBA or transient analysis.

3. The land use census is not utilized in any capacity as a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 309) of NEDO-31466, the land use census was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Land Use Census LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

6.3 INTERLABORATORY COMPARISON PROGRAM

LCO Statement:

Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the Commission. Participation in this program shall include all media for which samples are routinely collected and for which intercomparison samples are available.

Discussion:

The interlaboratory comparison program required by this specification confirms the accuracy of the measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public resulting from station operation. This program ensures independent checks on the precision and accuracy of the instrumentation used in the measurements of radioactive material for the radiological environmental monitoring program are performed.

Comparison to Screening Criteria:

1. The interlaboratory comparison program is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

2. The interlaboratory comparison program does not monitor a process variable that is an initial condition of a DBA or transient analysis.

3. The interlaboratory comparison program is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 309) of NEDO-31466, the interlaboratory comparison program was found to be a non-significant risk contributor to core damage frequency and offsite releases. NYPA has reviewed this evaluation, considers it applicable to JAFNPP, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the Interlaboratory Comparison Program LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

SECTION 3.3

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.3.1.1

Reactor Protection System (RPS) Instrumentation

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

MARKUP OF NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.3.1.1

Reactor Protection System (RPS) Instrumentation

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

Specification 3.3.1.1

A1

JAFNPP

1.1 FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective:

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Specifications:

A. Reactor Pressure > 785 psig and Core Flow > 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.09 shall constitute violation of the fuel cladding integrity safety limit, hereafter called the Safety Limit. An MCPR Safety Limit of 1.10 shall apply during single-loop operation.

See ITS: Chapter 2.0

2.1 FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

~~Specifications:~~ Table 3.3.1.1-1 Allowable Value

A19

A. ~~Trip Settings~~ Allowable Value

[Function] The limiting safety system ~~trip settings~~ shall be as specified below:

1. Neutron Flux ~~Trip~~ Settings

[1.a] a. IRM - The IRM flux scram ~~setting~~ shall be set at ≤120/125 of full scale.

A

Specification 3.3.1.1

A1

1.1 (cont'd)

See ITS' Chapter 2.0

B. Core Thermal Power Limit (Reactor Pressure ≤785 psig)

When the reactor pressure is ≤785 psig or core flow is less than or equal to 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

2.1 (cont'd)

Table 3.3.1.1-1 Allowable Value

A19

[Function] b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

[2.a] APRM - The APRM flux scram setting shall be ≤15 percent of rated neutron flux with the Reactor Mode Switch in Startup/Hot Standby ~~or Refuel~~. L4

Allowable Value

c. APRM Flux Scram Trip Settings (Run Mode)

[2.b]

(1) Flow Referenced Neutron Flux Scram Trip Setting

When the Mode Switch is in the RUN position, the APRM flow referenced flux scram trip setting shall be less than or equal to the limit specified in Table 3.1-1. This setting shall be adjusted during single loop operation when required by Specification 3.5.J.

A18

For no combination of recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 117% of rated thermal power.

Specification 3.3.1.1

1.1 (cont'd)

D. **Reactor Water Level (Hot or Cold Shutdown Conditions)**

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 18 inches above the Top of Active Fuel when it is seated in the core.

See ITS: Chapter 2.0

Table 3.3.1.1-1 Allowable Value

A19

A1

2.1 (cont'd)

[Functions]

[2.c]

(2) Fixed High Neutron Flux Scram Trip Setting

When the Mode Switch is in the RUN position, the APRM fixed high flux scram trip setting shall be:

$S' \leq 120\%$ Power

d. **APRM Rod Block Setting**

The APRM Rod block trip setting shall be less than or equal to the limit specified in Table 3.2-3. This setting shall be adjusted during single loop operation when required by Specification 3.5.J.

See ITS: 3.3.2.1

Specification 3.3.1.1

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Table 3.3.1.1-1 Allowable Value

A1

A17

2.1 (cont'd)

[Function]

[4] 2. Reactor Water Low Level Scram Trip Setting

Reactor low water level scram setting shall be ≥ 177 in. above the top of the active fuel (TAF) at normal operating conditions.

LA1

[8] 3. Turbine Stop Valve Closure Scram Trip Setting

Turbine stop valve scram shall be ≤ ~~10~~ 15 percent valve closure from full open when the reactor is at or above 29% of rated power.

L14

F

[9] 4. Turbine Control Valve Fast Closure Scram Trip Setting

Turbine control valve fast closure scram control oil pressure shall be set at 500 ≤ P ≤ 850 psig.

L14

F

[5] 5. Main Steam Line Isolation Valve Closure Scram Trip Setting

Main steam line isolation valve closure scram shall be ≤ 15 percent valve closure from full open.

6. Main Steam Line Isolation Valve Closure on Low Pressure

When in the run mode main steam line low pressure initiation of main steam line isolation valve closure shall be ≥ 825 psig.

See ITS 3.3.6.1

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Specification 3.3.1.1

A1

See ITS 2.0

1.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to limits on reactor coolant system pressure.

OBJECTIVE:

To establish a limit below which the integrity of the Reactor Coolant System is not threatened due to an overpressure condition.

SPECIFICATION:

1. The reactor vessel dome pressure shall not exceed 1,325 psig at any time when irradiated fuel is present in the reactor vessel.

Table 3.3.1.1-1 Allowable Value

A19

A1

2.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor coolant system safety limits from being exceeded.

OBJECTIVE:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

SPECIFICATION:

A19

1. The Limiting Safety System setting shall be specified below:

   [Function 3]

   A. Reactor coolant high pressure scram shall be $\leq$1,080 psig.

   B. At least 9 of the 11 reactor coolant system safety/relief valves shall have a nominal setting of 1,145 psig with an allowable setpoint error of $\pm$3 percent.

See ITS: 3.4.3

Specification 3.3.1.1

A1

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate the reactor scram.

Objective:

To assure the operability of the Reactor Protection System.

~~Specification:~~

[LCO 3.3.1.1] A. The setpoints and minimum number of instrument channels per trip system that must be operable for each position of the reactor mode switch, shall be as shown in [Table 3.3.1.1-1] Table ~~3.1-1~~. 3.3.1.1-1

4.1 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type of frequency of surveillance to be applied to the protection instrumentation.

~~Specification:~~ [Note 1 to SRs]

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1-1 and 4.1-2 respectively.

[SR 3.3.1.1.16] The response time of the reactor protection system trip functions listed below shall be demonstrated to be within its limit once per 24 months. [Note 1 to SR 3.3.1.1.16] Neutron detectors are exempt from response time testing. [Note 3 to SR 3.3.1.1.16] 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. MB

[Function]

[3] 1. Reactor High Pressure (02-3PT-55A, B, C, D)*
[6] 2. Drywell High Pressure (05PT-12A, B, C, D)
[4] 3. Reactor Water Level-Low (L3) (02-3LT-101A, B, C, D)*
[5] 4. Main Steam Line Isolation Valve Closure (29PNS-80A2, B2, C2, D2) (29PNS-86A2, B2, C2, D2)
[8] 5. Turbine Stop Valve Closure (94PNS-101, 102, 103, 104)
[9] 6. Turbine Control Valve Fast Closure (94PS-200A, B, C, D)
[2.c] 7. APRM Fixed High Neutron Flux
[2.b] 8. APRM Flow Referenced Neutron Flux

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LA14 * Sensor is eliminated from response time testing for the RPS actuation logic circuits. Response time testing and conformance to the test acceptance criteria for the remaining channel components includes trip unit and relay logic.

A14

RAI 3.3.1.1.6 and TSTF-332

B

Specification 3.3.1.1

A1

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TABLE 3.1-1 3.3.1.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Function No.	Required Minimum No. of Operable Instrument Channels Per Trip System (Notes 1 and 2)	Trip Function	Allowable Value Trip Level Setting	Applicable MODE or other specified condition — Mode in Which Function Must Be Operable: Refuel (Note 7) 5(a)	Startup 2	Run 1	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Conditions referenced from Required Action D.1 — Action (Note 3)
[10]	1	Mode Switch in Shutdown	N/A	X	X	X	1 Mode Switch	A — G, H (MODES 5 only)
[11]	1	Manual Scram	N/A	X	X	X	2	A
[1.a]	3	IRM High Flux	~~≤96% (120/125) of full scale~~	X	X		8	A
[1.b]	3	IRM Inoperative	N/A	X	X		8	A
[2.a]	2	APRM Neutron Flux-Startup ~~(Note 15)~~	≤15% Power	~~X~~	X		6	~~A~~ G
[2.b]	2	APRM Flow Referenced Neutron Flux ~~(Not to exceed 117%) (Note 13)~~	~~(Note 12)~~ As specified in the COLR			X	6	~~A or B~~ F
[2.c]	2	APRM Fixed High Neutron Flux	≤120% Power			X	6	~~A or B~~ F
[2.d]	2	APRM Inoperative	NA (Note 10)	~~X~~	X	X	6	G ~~A or B~~

A19 A3 A16 L1 L2 L3 L4 LA2 LA3 LA4 A4 A5 A1 M15 A3

Specification 3.3.1.1

A1

TABLE ~~3.1-1~~ 3.3.1.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Function No.	Required Minimum No. of Operable Instrument Channels Per Trip System (Notes 1 and 2)	Trip Function	Allowable Value Trip Level Setting (A19)	Applicable Mode or other Specified condition: Mode in Which Function Must Be Operable — Refuel (Note 7) (5(a))	Startup (2)	Run (1)	Total Number of Instrument Channels Provided by Design for Both Trip Systems (LA2)	Conditions referenced from Required Action D.1: Action (Note 3)
[3]	2	Reactor High Pressure	≤ ~~1060~~ 1079 psig (M16)	~~X~~ (Note 8) (A7) (L1)	X (A8)	X	~~4~~	L2 G ~~A~~
[6]	2	Drywell High Pressure (Note 16) (LA10)	≤ 2.7 psig	~~X~~ (Note 8)	X (Note 8)	X	~~4~~	L2 G ~~A~~
[4]	2	Reactor Low Water Level (Note 16)	≥ 177 in. above TAF (LA1)	~~X~~	X	X	~~4~~	L2 G ~~A~~ L2 L3 G, H (MODES only)
[7.a 7.b]	~~4~~ 2 2 (M2)	High Water Level in Scram Discharge Volume	≤ 34.5 gallons per Instrument Volume (Note 4) (M1)	X	X	X	~~8~~	L5 F ~~A~~
[5]	~~8~~ ~~8~~ (M3)	Main Steam Line Isolation Valve Closure	≤ 15% valve closure		LA5	X (Note 6)	~~8~~	
[9]	2	Turbine Control Valve Fast Closure	L14 500 ≤ P ≤ 850 psig ~~Control oil pressure between fast closure solenoid and disc dump valve~~ (LA6)			X (Note 5)	~~4~~	A9 ~~A or C~~ E
[8]	4	Turbine Stop Valve Closure	≤ ~~10~~ 15% valve closure (L14)			X (Notes 5 & ~~6~~) (LA5)	~~8~~	~~A or C~~ E

F

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A1

TABLE 3.1-1 (cont'd) 3.3.1.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

NOTES OF TABLE 3.1-1

add proposed Note to LCO 3.3.1.1 ACTIONS A2

[LCO 3.3.1.1]

1. There shall be two operable or tripped trip systems for each Trip Function, except as provided for below:

[ACTION A]

[RA A.1] a. For each Trip Function with one less than the required minimum number of operable instrument channels, place the inoperable instrument channel and/or its associated trip system in the tripped condition within 12 hours. Otherwise, initiate the ACTION required by Table 3.1-1 for the Trip Function. [RA A.2] [RA A.1] [ACTION D] 3.3.1.1-1

[ACTION B]

b. For each Trip Function with two or more channels less than the required minimum number of operable instrument channels:

[ACTION C]

(1) Within one hour, verify sufficient instrument channels remain operable or tripped* to maintain trip capability in the Trip Function, and LA13

[RA B.1] 2) Within 6 hours, place the inoperable instrument channel(s) in one trip system and/or that trip system** in the tripped condition* and

[RA B.2]

[ACTION A] (3) Within 12 hours, restore the inoperable instrument channel(s) in the other trip system to an operable status, or place the inoperable instrument channel(s) in the trip system and/or that trip system in the tripped condition*.

3.3.1.1-1

[ACTION D] If any of these three conditions cannot be satisfied, initiate the ACTION required by Table 3.1-1 for the affected Trip Function.

LA7

* An inoperable instrument channel or trip system need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, if the inoperable instrument channel is not restored to operable status within the required time, the ACTION required by Table 3.1-1 for that Trip Function shall be taken.

LA13

** This action applies to that trip system with the greatest number of inoperable instrument channels. If both systems have the same number of inoperable instrument channels, the ACTION can be applied to either trip system.

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

[Note 2 to SR's]

Specification 3.3.1.1

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3.3.1.1-1

TABLE ~~3.1-1~~ (cont'd)

REACTOR PROTECTION SYSTEM ~~(SCRAM)~~ INSTRUMENTATION ~~REQUIREMENTS~~

NOTES OF TABLE 3.1-1 (cont'd)

3. Action Statements:

12 L2

[ACTION G] A. Insert all operable control rods within four hours.
[ACTION F] B. Reduce power level to IRM range and place Mode Switch in the Startup position within eight hours.
[ACTION E] C. Reduce power level to less than 29 percent of rated within four hours.

4. Permissible to bypass, if the Reactor Mode Switch is in the Refuel or Shutdown position. M1

5. Bypassed when reactor power is less than 29 percent of rated power. [Applicability Functions 8 and 9]

6. The design permits closure of any two lines without a scram being initiated. LA5

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable: [footnote (a)]

[footnote (a)] A. Mode Switch in Shutdown.
B. Manual Scram.
C. High Flux IRM [Applicability for Function 7.a and 7.b]
L1
D. Scram Discharge Volume High Level when any control rod in a control cell containing fuel is not fully inserted.
E. APRM 15% Power Trip.
L4

8. Not required to be operable when primary containment integrity is not required. A8

9. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel. A7

10. An APRM will be considered operable if there are at least 2 LPRM inputs per level and at least 11 LPRM inputs of the normal complement. LA4

~~11. (Deleted)~~

B

Specification 3.3.1.1

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TABLE ~~3.1-1~~ 3.3.1.1-1 (cont'd)

REACTOR PROTECTION SYSTEM ~~(SCRAM)~~ INSTRUMENTATION ~~REQUIREMENTS~~

[2.6 Allowable Value]

Allowable Value A19

12. The APRM Flow Referenced Neutron Flux Scram ~~setting~~ shall be less than or equal to the limit specified in the Core Operating Limits Report.

LA3 Allowable Value A19

~~13. The Average Power Range Monitor scram function is varied as a function of recirculation flow (W).~~ The ~~trip setting~~ of this function must be maintained as specified in the Core Operating Limits Report.

~~14. Deleted.~~

LA3

~~15. This Average Power Range Monitor scram function is fixed point and is increased when the reactor mode switch is placed in the Run position.~~

~~16. Instrumentation common to PCIS.~~ LA10

Specification 3.3.1.1

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TABLE 4.1-1 3.3.1.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION TEST REQUIREMENTS

L6 A10 channel M4

	Trip Function	Group (Note 2)	Functional Test	Functional Test Frequency (Note 3)	Instrument Check
[10]	Mode Switch in Shutdown	A	Place Mode Switch in Shutdown (LA8)	R (SR 3.3.1.1.12)	NA (add proposed SR 3.3.1.1.14)
[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	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 3.3.1.1.14)	C	Trip Channel and Alarm (Note 4)	S/U and W (Note 5) (SR 3.0.4) (SR 3.3.1.1.3)	(L10) 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	NA
[2.b]	Flow Biased High Flux	B	Trip Output Relays (Note 4)	Q (SR 3.3.1.1.3)	NA (SR 3.3.1.1.6) (M5) (E)
[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	B	Trip Channel and Alarm (Note 4)	Q (A15)	D

Specification 3.3.1.1

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3.3.1.1-1

TABLE 4.1-1 (Cont'd)

M4

add proposed SR 3.3.1.1.14

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION TEST REQUIREMENTS

L6 A10 channel

Trip Function	Group (Note 2)	Functional Test	Functional Test Frequency (Note 3)	Instrument Check
[5] Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	SR 3.3.1.1.8 Q	NA
[9] Turbine Control Valve Fast Closure	A	Trip Channel and Alarm	Q	NA
[8][9] Turbine First Stage Pressure Permissive	B	Trip Channel and Alarm (Note 4)	Q	[8][9] D — 12 hours M6 SR 3.3.1.1
[8] Turbine Stop Valve Closure	A	Trip Channel and Alarm	Q	NA

LA12 L7 A12

NOTES FOR TABLE 4.1-1

LA9

1. The automatic scram contactors shall be exercised once every week by either using the RPS channel test switches or performing a functional test of any automatic scram function. If the contactors are exercised using a functional test of a scram function, the weekly test using the RPS channel test switch is considered satisfied. The automatic scram contactors shall also be exercised after maintenance on the contactors. L8

2. A description of the three groups is included in the Bases of this Specification. L6

3. Functional tests are not required on the part of the system that is not required to be operable or are tripped. If tests are missed on parts not required to be operable or are tripped, then they shall be performed prior to returning the system to an operable status. A10

A12 4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the instrument channels.

5. Weekly functional test required only during refuel and startup mode. — [1.a, 1.b and 2.a Applicable Modes]

6. The functional test shall be performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected.

LA11

Specification 3.3.1.1 (A1)

add proposed SR 3.3.1.1.13 (M9)

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TABLE 4.1-2 (3.3.1.1-1)

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

~~Instrument Channel~~ Function	Group (1)	Calibration	Frequency (2) (A10)
[1a] IRM High Flux	C	(M10) Comparison to APRM on Controlled Shutdowns	(L12) W ← [SR 3.3.1.1.6] ← add proposed Note (M14)
APRM High Flux Output Signal (add proposed SR 3.3.1.1.11 for Function 2.a, 2.b, 2.c) (M11)	B	[SR 3.3.1.1.2] ← add proposed Note; Heat Balance	~~D~~ 7 days (L13)
[2.b] Flow Bias Signal	B	Internal Power and Flow Test with Standard Pressure Source	R ← [SR 3.3.1.1.13]
[2.] (SR 3.3.1.1.7) LPRM Signal	B	[2.a, 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 3.3.1.1.13]
[6] High Drywell Pressure	B	Standard Pressure Source	(Note 6)
[4] Reactor Low Water Level	B	Standard Pressure Source	(Note 6)
[7.b] High Water Level in Scram Discharge Instrument Volume	A	Water Column ~~(Note 5)~~ (LA11)	R ~~(Note 5)~~ [SR 3.3.1.1.13]
[7.a] High Water Level in Scram Discharge Instrument Volume	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 3.3.1.1.13]
[8] [9] Turbine First Stage Pressure Permissive (LA12)	B	Standard Pressure Source	(Note 6) ← [SR 3.3.1.1.10 SR 3.3.1.1.13]

(L6) (L7)

add SR 3.3.1.1.6 to Function 2.a (M14)

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RAI 3.3.1.1-11

RAI 3.3.1.1-5

Specification 3.3.1.1

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TABLE ~~4.1-2 (Cont'd)~~ 3.3.1.1-1

REACTOR PROTECTION SYSTEM ~~(SCRAM)~~ INSTRUMENT ~~CALIBRATION~~

~~MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS~~

	FUNCTION ~~Instrument Channel~~	~~Group (1)~~	Calibration	Frequency ~~(2)~~ A10
[9]	Turbine Control Valve Fast Closure Oil Pressure Trip	~~A~~	~~Standard Pressure Source~~ L7	R — [SR 3.3.1.1.13]
[8]	Turbine Stop Valve Closure	~~A~~ L6	~~(Note 4)~~	~~(Note 4)~~ [SR 3.3.1.1.13]

add SR 3.3.1.1.15 — M13

NOTES FOR TABLE 4.1-2

1. ~~A description of three groups is included in the Bases of this Specification.~~

A10 — 2. ~~Calibration test is not required on the part of the system that is not required to be operable, or is tripped, but is required prior to return to service.~~

3. Deleted

4. ~~Actuation of these switches by normal means will be performed once per 24 months.~~ M12

LA11 — 5. ~~Calibration shall be performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected.~~

6. Sensor calibration once per 24 months. Master/slave trip unit calibration once per 6 months.

[SR 3.3.1.1.13] [SR 3.3.1.1.10]

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.3.1.1

Reactor Protection System (RPS) Instrumentation

DISCUSSION OF CHANGES (DOCs) TO THE CTS

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted 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 This change proposes to add ITS 3.3.1.1 ACTIONS Note, consistent with conditional details ("For each Trip Function...") contained in CTS Table 3.1-1 Note 1.a, which will allow separate Condition entry for each channel. In conjunction with proposed Specification 1.3 - "Completion Times," the Note ("Separate condition entry ...") and the Conditions of ITS 3.3.1.1 provide more explicit direction of the current interpretation of the existing Specifications. This change in presentation method provides instructions, in a manner more explicit for proper application of the Actions for Technical Specification compliance, consistent with the format and requirements of NUREG-1433, Revision 1. Therefore, this change is considered administrative.

A3 The Trip Level Settings in CTS Table 3.1-1 for the Mode Switch in Shutdown, Manual Scram, IRM Inoperative and APRM Inoperative Functions have been changed to NA, since in reality, there are no Allowable Values. These Functions are the result of mechanically actuated contacts or are dependent on fixed configurations and are not adjustable (i.e., the setpoints cannot be adjusted). Since CTS Table 3.1-1 does not specify Trip Level Settings for these Functions this change is considered administrative.

A4 The Actions in CTS Table 3.1-1 for APRM Flow Referenced Neutron Flux provides an option of either Action A, inserting all Operable rods within 4 hours (being in MODE 3), or Action B reducing power to the IRM range and placing the reactor mode selector switch in startup (being in MODE 2) within eight hours if the APRM Flow Referenced Neutron Flux Function has less than the minimum number of Operable channels per trip system. Proposed ITS 3.3.1.1 ACTION F requires entry into MODE 2. The APRM Flow Referenced Neutron Flux Function is only required in MODE 1 when there is a possibility of generating excessive THERMAL POWER and potentially exceeding the Safety Limit applicable to high pressure and core flow conditions (MCPR Safety Limit). During Modes 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity. Therefore, once the plant reaches MODE 2, the LCO is no longer applicable. The CTS option of proceeding to MODE 3 is unnecessary and would not be used; therefore, this change is considered administrative.

ADMINISTRATIVE CHANGES

A5 The Actions in CTS Table 3.1-1 for APRM Fixed High Neutron Flux provides an option of either Action A, inserting all Operable rods within 4 hours (being in MODE 3), or Action B reducing power to the IRM range and placing the reactor mode selector switch in startup (being in MODE 2) within 8 hours if the APRM Fixed High Neutron Flux Function has less than the minimum number of Operable channels per trip system. Proposed ITS 3.3.1.1 ACTION F requires entry into MODE 2. The APRM Fixed High Neutron Flux Function is only required in MODE 1 where the potential consequences of the analyzed transients could result in the Safety Limits (e.g., MCPR and RCS Pressure) being exceeded. During Modes 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity. Therefore, once the plant reaches MODE 2, the LCO is no longer applicable. The CTS option of proceeding to MODE 3 is unnecessary and would not be used; therefore, this change is considered administrative.

A6 Not Used.

A7 This change proposes to delete the following requirements in CTS Table 3.1-1 for the RPS Functions when in Mode 5.

The High Reactor Pressure Function will be Operable with the mode switch in refuel and the reactor pressure vessel head bolted to the vessel.

The High Drywell Pressure Function will be Operable with the mode switch in refuel and primary containment integrity required.

The Reactor Low Water Level Function will be Operable with the mode switch in refuel.

The proposed change will delete the requirement for these Functions to be Operable when the mode switch is in the refuel position (even if rods are withdrawn). The High Reactor Pressure function is not required in MODE 5 because the RCS is not pressurized and the reactor pressure vessel head is not bolted on (see Note 9 allowance). The High Drywell Pressure Function is not required in MODE 5 because there is not enough energy in the RCS to overpressurize the drywell and containment integrity is not required. The Reactor Low Water Level Function is not required in MODE 5 because proposed Specifications 3.9.6, "RPV Water Level, "3.9.7, "RHR - High Water Level, " 3.9.8, "RHR - Low Water Level," ensure adequate cooling and retention of fission product activity. These changes are consistent with NUREG-1433, Revision 1, and are considered administrative because Note (7) to Table 3.1-1 states that in this condition (effectively MODES 4 and 5) only the Mode Switch in

ADMINISTRATIVE CHANGES

A7 (continued)

Shutdown Function, Manual Scram Function, High Flux IRM Function, Scram Discharge Volume High Level Function (when any control rod in any cell containing fuel is not fully inserted) and APRM High Flux (Startup) Functions need be OPERABLE.

A8 CTS Table 3.1-1 Note 8, that permits the High Drywell Pressure Function to be inoperable when primary containment integrity is not required in the Refuel and Startup Modes, is being deleted. Primary containment integrity, via the proposed Specifications of Section 3.6, is required in MODE 2. In addition, the requirement for High Drywell Pressure Function in Refuel (MODE 5) has been deleted (see A7). Consequently, the Note which allows this Function to be inoperable in Startup (MODE 2) and Refuel (MODE 5) when primary containment integrity is not required has been deleted. Therefore, deletion of this Note is considered to be administrative and is consistent with NUREG-1433, Revision 1.

A9 The Actions in CTS Table 3.1-1 for when the Turbine Control Valve Fast Closure or the Turbine Stop Valve Closure Functions have less than the minimum number of Operable channels per trip system provides an option, Action A, to either insert all Operable rods within 4 hours (being in MODE 3) or, Action C, reducing power to less than 29% of Rated Power. ITS 3.3.1.1 ACTION E requires only that Thermal Power be reduced to less than 29% RTP. These Functions are not required when Thermal Power is < 29% RTP since the Reactor Vessel Pressure – High and the Average Power Range Monitor Neutron Flux – High (Fixed) Functions are adequate to maintain the necessary safety margins. Consequently, once Thermal Power is < 29% RTP, the LCO is no longer applicable and the CTS option of proceeding to MODE 3 is unnecessary and would not be used since the option to shutdown always exists. Therefore, this change is considered administrative.

A10 CTS Table 4.1-1 Note 3 states that functional tests are not required on the part of the system that is not required to be operable or are tripped. If tests are missed on parts not required to be operable or are tripped, then they shall be performed prior to returning the system to an operable status. This explicit requirement is not retained in ITS 3.3.1.1. This explicit Note is not needed in ITS 3.3.1.1 since these allowances are included in ITS SR 3.0.1. SR 3.0.1 states that SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. In addition, the Note states that Surveillances do not have to be performed on inoperable equipment or variables outside specified limits. When equipment is declared inoperable, the Actions of this LCO require

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A10 (continued)

the equipment to be placed in the trip condition. In this condition, the equipment is still inoperable but has accomplished the required safety function. Therefore the allowances in SR 3.0.1 and the associated actions provide adequate guidance with respect to when the associated surveillances are required to be performed and this explicit requirement is not retained. CTS Table 4.1-2 Note 2 contains a similar note and it is also deleted. This change is consistent with NUREG-1433, Revision 1.

A11 Not Used.

A12 CTS Table 4.1-1 Note 4 specifies that certain instrumentation is excepted from the instrumentation channel test definition. The instrumentation channel functional test will consist of injecting a simulated electrical signal into the instrument channels. This explicit allowance is not retained in ITS 3.3.1.1 since it is duplicative of the current and proposed CHANNEL FUNCTIONAL TEST definition in ITS Chapter 1.0. Since this change does not change any technical requirements, it is considered administrative. This change is consistent with NUREG-1433, Revision 1.

A13 Not Used.

A14 CTS 4.1.A Note * specifies that Response Time Testing and conformance to the test acceptance criteria for the remaining channel components includes trip unit and relay logic. This requirement is not explicitly included in ITS SR 3.3.1.1.16 since the definition of RPS RESPONSE TIME in ITS Chapter 1.0 and SR 3.3.1.1.16 ensure the proper testing is performed. Since this deletion does not change any current requirements, this change is considered administrative. RAI 3.3.1.1-6

A15 The explicit requirement to perform a quarterly Functional Test of the High Water Level in Scram Discharge Instrument Volume Function of CTS Table 4.1-1 is being deleted. CTS Table 4.1-2 and ITS SR 3.3.1.1.9 require a CHANNEL CALIBRATION at the same Frequency, therefore this explicit requirement to perform a quarterly CHANNEL FUNCTIONAL TEST is not required since the ITS definition of CHANNEL CALIBRATION fulfills all the requirements of the CHANNEL FUNCTIONAL TEST. This change is considered administrative since the existing requirements will be fulfilled by performing a CHANNEL CALIBRATION every 92 days.

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A16 CTS Table 3.3-1 specifies that the trip level setpoint of the IRM High Flux Function is ≤ 96% (120/125) of full scale. The Allowable Value retained for this Function (ITS Table 3.3.1.1-1 Function 1.a) in the ITS is ≤ 120/125 divisions of full scale. Since the current and proposed values are equivalent, this change is considered administrative. This change in format is consistent with NUREG-1433, Revision 1.

A17 CTS Table 4.1-1 specifies that a Functional test of the RPS Channel Test switches are required to be performed on a weekly (W) basis. Note 1 of the Table clarifies that this test is to exercise the automatic scram contactors by either the RPS channel test switches or by performing a functional test of any automatic scram function. Therefore, ITS SR 3.3.1.1.4 requires the performance of a functional test of the automatic scram contactors every 7 days. Since this change does not change any technical requirements this change is considered administrative. The details of CTS Table 4.1-1 Note 1 have been relocated to the Bases according to LA9.

A18 CTS 2.1.A.1.c specifies that the APRM Flow Referenced Neutron Flux Scram Trip Setting shall be adjusted during single loop operation when required by Specification 3.5.J (The actual requirement is specified in CTS 3.5.K). This cross reference is deleted since the explicit requirement that the Allowable Values must be adjusted is included in proposed ITS LCO 3.4.1.c. This cross reference is included in ITS 3.4.1, "Recirculation Loops Operating" since this Specification provides the specific requirements that must be met for single loop operation. The actual Allowable Values are included in the COLR since the values are fuel cycle dependent. Since the ITS will continue to require this adjustment, this change is considered administrative.

A19 CTS Table 3.1-1 includes a "Trip Level Setting" column. The setting for each Reactor Protection System (RPS) Function is listed in this column. In some cases the settings are also duplicated in CTS 2.1.A (Fuel Cladding integrity - Trip Settings). This CTS Section also refers to these settings as the "limiting safety system trip settings" consistent with the terminology used in 10 CFR 50.36. In the ITS, the RPS Functions are included in Table 3.3.1.1-1 along with its associated "Allowable Value".

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

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A19 (continued)

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

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

The setpoints specified in the setpoint calculations are selected to ensure that the actual field trip setpoints do not exceed the ITS Allowable Values (i.e., the CTS "trip level settings" and the CTS "trip settings") between successive CHANNEL CALIBRATIONS. The CTS "trip settings"/"trip level settings" and the "ITS Allowable Values" are both the TS limit values that are placed on the actual field setpoints. The Allowable Values are derived from the trip setpoints by accounting for normal effects that would be seen during periodic surveillance or calibration. These effects are instrumentation uncertainties observed during normal operation (e.g., drift and calibration uncertainties). Accordingly, the ITS Allowable Values include all applicable instrument channel and measurement uncertainties. A channel is inoperable if its actual field trip setpoint is not within its required ITS Allowable Value. RAI 3.3.1.1-1

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

These "Trip Level Settings" or "Allowable Values" have been established consistent with the NYPA Engineering Standards Manual, IES-3A, "Instrument Loop Accuracy and Setpoint Calculation Methodology." The methodology used to determine the "Allowable Values" are consistent with the methodology discussed in ISA-S67.04-1994, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." This change revises the terminology used in the CTS from "Trip Level Setting" or "limiting safety system trip settings" to LAI

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.

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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 Instrument 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.

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.

TECHNICAL CHANGES - MORE RESTRICTIVE

M4 ITS SR 3.3.1.1.14 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
SDIV Water Level – High (MODE 1, MODE 2, and MODE 5(a))
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)

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.

TECHNICAL CHANGES - MORE RESTRICTIVE

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)
High Water Level in Scram Discharge Instrument Volume (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 withdrawing SRMs from the fully inserted position. 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.

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 3.3.1.1.16 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 3.3.1.1.16 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 instead of one channel in each trip system required by the CTS. This change will ensure a sufficient number of channels are tested each interval to identify any significant response time degradation.

M9 ITS SR 3.3.1.1.13 is added to CTS Table 4.1-2 to perform a Channel Calibration of the IRM Neutron Flux – High Function (MODE 2 and

TECHNICAL CHANGES - MORE RESTRICTIVE

M9 (continued)

MODE 5(a)) every 24 months. The addition of new requirements (Surveillances) to the current Technical Specifications constitute a more restrictive change necessary to ensure the RPS Functions are maintained Operable. In addition, two Notes have been added along with this surveillance to clarify that: (1) the neutron detectors are excluded from the calibration and, (2) the calibration is not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. This change is consistent with NUREG-1433, Revision 1.

M10 CTS Table 4.1-2 requires only a heat balance for APRM High Flux Output Signal calibration. ITS SR 3.3.1.1.2 additionally requires that the absolute difference between the APRM channels and the calculated power be ≤ 2% RTP plus any gain adjustment required by LCO 3.2.3, "Average Power Range Monitor (APRM) Gain and Setpoint" while operating at ≥ 25% RTP (L10). The addition of acceptance criteria to ensure instrument OPERABILITY constitutes a more restrictive change. The requirement to adjust the gain in accordance with LCO 3.2.4 is consistent with current practice in CTS 4.1.B. This change is consistent with NUREG-1433, Revision 1. This change is not considered to result in any reduction to safety.

M11 A new requirement has been added (ITS SR 3.3.1.1.11) to perform a Channel Calibration, of the APRM Neutron Flux—High (Startup) (MODE 2), APRM Neutron Flux—High (Flow Biased) and APRM Neutron Flux—High (Fixed) Functions, every 184 days. This 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.

M12 CTS Table 4.1.2, Note 4 requires actuation of the MSIV Closure limit switches and Turbine Stop Valve Closure pressure switches by normal means every 24 months. ITS SR 3.3.1.1.13 requires an actual Channel Calibration of these instruments every 24 months to ensure channel OPERABILITY. This change in 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.

TECHNICAL CHANGES - MORE RESTRICTIVE

M13 A new requirement has been added (ITS SR 3.3.1.1.15) to the Surveillances of CTS Table 4.1-2 to verify the Turbine Stop Valve – Closure and Turbine Control Valve Fast Closure, EHC Oil Pressure – Low Functions are not bypassed when THERMAL POWER is ≥ 29 % RTP at a Frequency of 24 months. The addition of new requirement (Surveillance) to the current Technical Specifications constitutes a more restrictive change necessary to ensure the associated RPS Functions are maintained Operable when required. This change is consistent with NUREG-1433, Revision 1.

M14 CTS 4.1-2 requires a comparison of the IRM channels with the APRM channels on a controlled shutdown. However, the requirement is only associated with the IRM High Flux Function in the CTS. In the ITS, this test (ITS SR 3.3.1.1.6) is associated with ITS Table 3.3.1.1-1 Functions 1.a (IRM Neutron Flux-High) and 2.a (APRM Neutron Flux-High (Startup)) since it is equally important to both Functions and the explicit requirement is to verify the IRM and APRM channels overlap. In addition, a Note is included which states that the SR is only required to be met during entry into MODE 2 from MODE 1 since this is when the IRM and APRM channels are designed to overlap with one another. Currently, the Surveillance implies that the calibration is to be performed on controlled shutdowns. It does not imply that the Surveillance is required to be met during the entire shutdown. The overlap can not exist during the entire shutdown since the APRMs may be reading downscale during operations in MODE 2. Since the requirement is more explicit to when the requirement must be met and since the association is related to both of the specified Functions this change is considered more restrictive on plant operations. This change is consistent with NUREG-1433, Revision 1.

M15 The Actions in CTS Table 3.1-1 for the APRM Inoperative Function provides an option of either Action A, inserting all Operable rods within 4 hours (being in MODE 3), or Action B reducing power to the IRM range and placing the reactor mode selector switch in startup (being in MODE 2) within 8 hours if the APRM Inoperative Function has less than the minimum number of Operable channels per trip system. ITS 3.3.1.1 ACTION G requires entry into MODE 3 since the APRM Inoperative Function is required in MODEs 1 and 2. CTS Table 3.1-1 requires the Function to be OPERABLE in startup (MODE 2), the Action B option of reducing power

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M15 (continued)

and placing the reactor mode switch in startup (MODE 2) will not place the plant outside of the associated Applicability (MODE 1 and 2). This allowance is not consistent with the philosophy of the ITS, since it does not place the plant outside the Applicability of the Specification. Therefore, this option has been deleted. Since the option has been deleted this change is considered more restrictive on plant operation, but necessary to ensure proper actions are taken when the APRM Inoperative Function is inoperable. The proposed Action is consistent with the default actions for the APRM Neutron Flux - High (Startup) which also has an Applicability of MODE 2. This change is consistent with NUREG-1433, Revision 1.

M16 This change replaces the setpoint or Allowable Value (A19) in CTS Table 3.1-1, Item 9, Reactor High Pressure ≤ 1080 psig with ≤ 1079 psig (ITS Table 3.3.1.1-1, Function 3, Reactor Pressure – High). 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." The proposed value will ensure the most limiting requirement is met. All design limits, applied in the methodologies, were confirmed as ensuring that applicable design requirements of the associated system is maintained. F

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The detail in CTS 2.1.A.2 and CTS Table 3.1-1 that the Trip Level Setting of the Reactor Low Water Level Function is referenced from the Top of Active Fuel (TAF) is proposed to be relocated to the Bases. CTS 1.0.Z definition specifies that the Top of Active Fuel, corresponding to the top of the enriched fuel column of each fuel bundle, is located 352.5 inches above vessel zero, which is the lowest point in the inside bottom of the reactor pressure vessel. (See General Electric drawing No. 919D690BD). These details are also proposed to be relocated to the Bases. The requirement in ITS LCO 3.3.1.1 that the RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE, the requirements in the Table including the Allowable Value for the Reactor Water Level – Low (Level 3) Function, the definition of Operability, the proposed Actions, and Surveillance Requirements are adequate to ensure

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 (continued)

the instrumentation is properly maintained. In addition, the Bases includes a statement that the Reactor Water Level – Low (Level 3) Allowable Value The Allowable Value corresponds to a level of water 352.56 inches above the lowest point in the inside bottom of the reactor pressure vessel and also corresponds to the top of a 144 inch fuel column. 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.

LA2 CTS Table 3.3-1 Column "Total Number of Instrument Channels Provided by Design for Both Trip Systems", is to be relocated to the Bases. These details related to system design are not necessary to ensure the associated instruments remain OPERABLE. The requirements of ITS 3.3.1.1 which require the RPS instrumentation to be OPERABLE and the definition of OPERABILITY suffice. Therefore, the relocated 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.

LA3 The details in CTS Table 3.1.1 Note 15, stating this Average Power Range Monitor (APRM) scram function is fixed point and is increased when the reactor mode switch is placed in the Run position, and the details in Note 13, stating the APRM Flow Referenced Neutron Flux scram function is varied as a function of recirculation flow (W) is proposed to be relocated to the Bases. These are informational Notes which describe the design of the instrumentation, and which are not needed to comply with Technical Specifications. These details related to system operation are not necessary to ensure the associated instruments remain Operable. The requirements of ITS 3.3.1.1 which require the RPS instrumentation to be OPERABLE and the definition of OPERABILITY suffice. Therefore, the relocated 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.

LA4 This change proposes to relocate the requirement contained in Note 10 of CTS Table 3.1-1, that an APRM will be considered inoperable if there are less than 2 LPRM inputs per level or less than 11 operable LPRM detectors to an APRM, to the Bases. The details for system Operability are not necessary to ensure the APRMs are OPERABLE. The requirements of ITS 3.3.1.1 which require the RPS instrumentation to be OPERABLE and the

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA4 (continued)

definition of OPERABILITY suffice. Therefore, the relocated 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 ITS.

LA5 CTS Table 3.1-1 Note 6, statement regarding the function's design which permits closure of any two lines without a scram being initiated, is proposed to be relocated to the Bases. The details of system design are not necessary to ensure the MSIV-Closure instrumentation is OPERABLE. The requirements of ITS 3.3.1.1 which require the MSIV-Closure instrumentation to be OPERABLE and the definition of OPERABILITY suffice. Therefore, the relocated 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.

LA6 The design detail in CTS Table 3.1-1 Turbine Control Valve Fast Closure, Trip Level Setting, regarding the physical location of the pressure switch, is proposed to be relocated to the Bases. The details of systemdesign are not necessary to ensure the Turbine Control Valve Fast Closure instrumentation is OPERABLE. The requirements of ITS 3.3.1.1 which require the Turbine Control Valve Fast Closure instrumentation to be OPERABLE and the definition of OPERABILITY suffice. Therefore, the relocated 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.

LA7 The details contained in CTS Table 3.1-1, Notes of Table 3.1-1 footnote *, providing conditions and precautions for placing an inoperable channel or trip system in trip, are to be relocated to the Bases. These details related to system configuration are not necessary to ensure the associated instruments remain OPERABLE. The requirements of ITS 3.3.1.1 which require the RPS instrumentation to be OPERABLE and the definition of OPERABILITY suffice. Therefore, the relocated 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.

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LA8 The details contained in CTS Table 4.1-1 Mode switch in Shutdown Functional Test Requirements, for the performance of the Channel Functional Test of the Mode Switch in Shutdown which requires placing the Mode Switch in Shutdown, is being relocated to the Bases. The details for system OPERABILITY are not necessary to ensure the Reactor Mode Switch - Shutdown Position Function 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.

LA9 The details contained in CTS Table 4.1-1 Note 1, allowing exercising of the automatic scram contactors by performing a functional test of an automatic scram function or using the RPS Channel Test Switch, are being relocated to the Bases. The details for system OPERABILITY 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.

LA10 The design details contained in CTS Table 3.1-1 Note 16, that state the instrumentation (Drywell High Pressure and Reactor Low Water Level) are common to PCIS, are proposed to be relocated to the UFSAR. These design details are not necessary to be included in the Technical Specifications to ensure the OPERABILITY of the RPS instrumentation. 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 UFSAR will be controlled by the provisions of 10 CFR 50.59.

LA11 Details of the methods in CTS Table 4.1-1 Note 6 and Table 4.1-2 Note 5, that require testing using a water column or similar device to provide assurance that damage to a float or other portion of the float assembly will be detected, is being relocated to the Bases. The details for performing system Operability are not necessary to ensure the High Water Level Scram Discharge Instrument Volume Function 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.

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LA12 CTS Tables 4.1-1 and 4.1-2 identify the Turbine First Stage Pressure Permissive as a separate Function. ITS Table 3.3.1.1-1 includes the current Turbine First Stage Pressure Permissive Surveillances in the Surveillances for Function 8, Turbine Stop Valve - Closure and for Function 9, Turbine Control Valve Fast Closure, EHC Oil Pressure - Low. Testing of the Turbine First Stage Pressure Permissive is included in ITS SR 3.3.1.1.15 (see M13). This change proposes to relocate the listing of this Function from CTS Tables 4.1.1 and 4.1.2 to the Bases for proposed Functions 8 and 9 and SR 3.3.1.1.15. The identification of the Turbine First Stage Pressure Permissive as a separate Function is not necessary to ensure the instrumentation remains Operable. The requirements of ITS 3.3.1.1 Functions 8 and 9 which require the Turbine First Stage Pressure Permissive 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.

LA13 The operational details in CTS Table 3.1-1 Notes, footnote **, that state that the trip system with the greatest number of inoperable instrument channels should be the trip system that is tripped, is being 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.

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.

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 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 Action 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,

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 (continued)

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 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).

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 (continued)

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).

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)

RAI 3.3.1.1-2

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.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 (continued)

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 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

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 (continued)

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 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.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 (continued)

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.

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.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L5 The CTS Table 3.3-1 Action 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 8 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. 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 Completion Time of 8 hours to be in MODE 2 is acceptable due to the low probability of an event requiring this Function during the proposed additional 4 hours. In addition, the 8 hour Completion Time provides sufficient time to reach MODE 2 without challenging plant systems. This change is consistent with NUREG-1433, Revision 1.

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

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L8 (continued)

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.

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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 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

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L12 (continued)

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. This change is consistent with NUREG-1433, Revision 1.

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. E

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.3.1.1

Reactor Protection System (RPS) Instrumentation

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

TECHNICAL CHANGE - 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?

   This change will require the associated RPS Functions (Mode Switch in Shutdown, Manual Scram, IRM Neutron Flux—High, and IRM Inoperable) to be Operable when in MODE 5 only with any control rod withdrawn from a core cell containing one or more fuel assemblies. The proposed change does not affect the probability of an accident. These Functions are not assumed in the accident analysis when in MODE 5 with all control rods inserted in core cells containing one or more fuel assemblies. The design function of these RPS Functions is to shutdown the reactor when required by initiating a reactor scram. This is only possible when control rods are withdrawn. With all the control rods inserted, the shutdown margin requirements and the required one-rod-out interlock ensure no positive reactivity excursion will occur. This change will continue to ensure the RPS Instrumentation is maintained consistent with analysis assumptions. The consequences of an accident are not affected by this change. This change will not alter assumptions relative to the mitigation of an accident or transient event. 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?

   This change will require the associated RPS Functions to be Operable only when in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. The proposed change to the Applicability will not create the possibility of an accident. This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

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

   This change will require the associated RPS Functions to be Operable only when in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. The margin of safety will not be affected by this change. The design function of the RPS Functions is to shutdown the reactor by initiating a reactor scram. This is only possible 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 and the required one-rod-out interlock ensure no event will occur. The safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGE - 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?

   This change allows an additional 8 hours to reach MODE 3 when a Required Action and associated Completion Time are not met. This provides a reasonable amount of time to perform an orderly shutdown, thus minimizing a potential upset from a too rapid decrease in plant power. The probability of an accident is not increased because the RPS Instrumentation is not assumed to be an initiator of any analyzed event. The change will not allow continuous operation such that a single failure will preclude the affected channel's function from being performed. The consequences of an event occurring while the plant is being shutdown during the extra 8 hours are the same as the consequences of an event occurring for the current 4 hours. 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 proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it 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 increased time allowed for reaching MODE 3 with inoperable RPS channels is acceptable based on the small probability of an event requiring the inoperable channels to function and the desire to minimize plant transients. The requested 8 hour extension will provide sufficient time for the plant to reach MODE 3 in an orderly manner. As a result, the potential for human error will be reduced.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

3. (continued)

Any reduction in a margin of safety will be insignificant and offset by the benefit gained from providing sufficient time to reach MODE 3, thus avoiding potential plant transients from attempting to reach MODE 3 in the current time.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L3 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 will require only the control rods in core cells containing one or more fuel assemblies to be inserted if a Reactor Protection System (RPS) Function is inoperable and RPS trip capability cannot be restored in the specified Completion Time. The probability of an accident is not increased by this change because the insertion of control rods in response to the inability to satisfy Required Actions is not considered the initiator of any analyzed event. The consequences of an accident will not be increased because a core cell without any fuel assemblies and with the associated control rod fully withdrawn contributes less reactivity to the core than a core cell with one or more fuel assemblies and a fully inserted control rod. As a result, the absence of all four fuel assemblies satisfies the safety objective of fully inserting a control rod. 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?

   This proposed change will not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, modified, tested, or inspected. Therefore, this 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 proposed change will require only the control rods in core cells containing one or more fuel assemblies to be inserted if a Reactor Protection System (RPS) Function is inoperable and RPS trip capability cannot be restored in the specified Completion Time. The proposed change does not involve a significant reduction in a margin of safety

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

3. (continued)

because a core cell without any fuel assemblies and with the associated control rod fully withdrawn contributes less reactivity to the core than a core cell with one or more fuel assemblies and a fully inserted control rod. As a result, the absence of all four fuel assemblies satisfies the safety objective of fully inserting a control rod. As a result, the change does not affect the current analysis assumptions. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGE - 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?

   Not requiring APRMs to be operational in the Refuel Mode will not increase the probability of inadvertent reactor criticality during refueling operations. RIs, NMS (SRMs, IRMs), and procedural restrictions provide assurance that inadvertent criticality does not occur due to the simultaneous withdrawal or removal of two control rods (except as permitted by TS section 3.10, "Core Alterations" and ITS 3.10.6, "Multiple Control Rod Withdrawal - Refueling") or due to the inadvertent insertion of a fuel assembly into a core location with a control rod removed or withdrawn.

   The FSAR Section 14.5.4 discusses the potential for a control rod withdrawal error during refueling and startup operations. The discussion concludes that the withdrawal of one control rod does not require a safety action because the total worth of one control rod is not sufficient to cause criticality. The attempted withdrawal of two control rods in fuel cells containing fuel, assuming an operator error and a single active failure, would result in a control rod block initiated by the RIs. The safety-related IRM subsystem, which is required by TS to be operable while in the Refuel Mode, is designed to generate a reactor scram on high neutron flux and is therefore a backup protective system for the RIs during refueling.

   The safety-related IRM subsystem of the NMS is required by TS to be operable during refueling to support the safety design bases of the NMS and RPS. The SRM is not a safety-related subsystem but is important to plant safety and is required by TS to be operable in the Refuel Mode. The SRM subsystem provides the plant operator with indication of neutron flux levels from startup conditions to the IRM operating range. The SRMs and IRMs are designed to respond to local core conditions and would indicate and respond (a scram) to an accident condition to mitigate the transient. Thus, the APRMs are not necessary in the Refuel Mode. The

RAI 3.3.1.1-2

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L4 CHANGE

1. (continued)

proposed TS change will not alter the ITS requirements that the APRMs be operable during shutdown margin demonstrations, because the reactor will be in Mode 5 with the Mode switch in the Startup/Hot Standby position in accordance with ITS 3.10.8, "SDM Test - Refueling."

The proposed TS change eliminates the APRM operability requirement when in the Refuel mode and would not affect the UFSAR evaluation of the inadvertent criticality due to the withdrawal or removal of the highest worth control rod or due to the insertion of fuel assemblies in uncontrolled cells. The UFSAR concludes that the RIs and plant procedures provide assurance that inadvertent criticality does not occur during refueling.

The consequences of an accident will not be increased by the proposed TS change because of the existing lines of defense which prevent an inadvertent criticality event during refueling, e.g., administrative restrictions, refueling procedures, licensed plant operators, SRMs, RIs, and IRMs. Furthermore, should the number of operable IRM or SRM channels be less than that required by TS, the TS require that core alteration activities be suspended and/or all insertable control rods be inserted into core cells containing one or more fuel assemblies (ITS 3.3.1.1 & 3.3.1.2). RAI 3.3.1.1-2

Therefore, the proposed changes do not result in an 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 changes to the TS will remove the APRM operability requirement when in the Refuel Mode; however, the SRMs and IRMs will still be required to be operable. The IRMs are safety-related and are designed to detect and respond to increases in neutron flux within the local core regions. Any increases in neutron flux during refueling would originate at a local core location, i.e., due to rod withdrawal or fuel assembly insertion. TS require IRM operability and IRMs will generate an RPS scram if neutron flux increases to the setpoint. Therefore, removing the APRM operability requirement in the Refuel Mode would not effect any safety-related equipment or equipment important to safety. RAI 3.3.1.1-2

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L4 CHANGE

2. (continued)

Removing the APRM operability in the Refuel Mode will not affect the response of safety-related equipment as previously evaluated in the FSAR. The proposed changes to the TS do not affect any safety-related equipment or equipment important to safety, other than the APRMs.

No new types of accidents would be introduced since the SRMs and IRMs are required to be operable in the Refuel Mode. Both SRMs and IRMs monitor neutron flux. The IRMs would provide a scram signal, as appropriate, in respoonse to an increase in neutron flux to mitigate a transient event. Furthermore, should the number of operable IRM or SRM channels be less than that required by TS, the TS require that core alteration activities be suspended and/or all insertable control rods be inserted into core cells containing one or more fuel assemblies (ITS 3.3.1.1 & 3.3.1.2).

Finally, the APRMs do not have functions which can cause an accident condition.

Therefore, the proposed TS changes do 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?

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 removal or withdrawal of more than one control rod and prevent the insertion of fuel assemblies into the core unless the control rod for the

RAI 3.3.1.1-2

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L4 CHANGE

4. (continued)

applicable fuel cell is fully inserted (except as permitted by TS section 3.10, "Core Alterations" and ITS 3.10.6, "Multiple Control Rod Withdrawal - Refueling").

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 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. For those events in which an APRM trip would occur, the IRM trip would occur at much lower flux levels. A reactivity insertion transient that could occur 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 that would result in a local core criticality transient would be 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.

Therefore, the proposed TS changes do not involve a significant reduction in a margin of safety.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L5 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 will relax the current Required Actions for the Main Steam Isolation Valve Closure Function whenever an inoperable channel or trip system cannot be placed in trip within the required Completion Time. The current Actions require the rods to be inserted within 4 hours. The proposed change will require the plant to be brought to MODE 2 within 8 hours. The probability of an accident is not increased by this change because the change does not involve activities assumed to be initiators of any analyzed event. The consequences of an accident will not be increased because: the MSIV Closure Function is only required in MODE 1 when, with the MSIVs open and the heat generation rate high, a pressurization transient can occur if the MSIVs close. In MODE 2, the heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection. 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?

   This proposed change will not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, modified, tested, or inspected. Therefore, this change will not create the possibility of a new or different kind of accident previously evaluated.

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

   The proposed change does not result in a significant reduction in the margin of safety because: the change does not involve changes to any plant hardware or plant operating procedures; the change in the proposed Required Actions does not involve activities assumed to be initiators of any analyzed event; placing the reactor in MODE 2 versus inserting all control rods is sufficient to ensure that the heat generation rate is

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L5 CHANGE

3. (continued)

low enough that the other diverse RPS functions provide adequate protection; and, the change will not allow continuous operation with plant conditions such that a single failure will preclude the scram function from being performed. In addition, the Completion Time of 8 hours to be in MODE 2 is acceptable due to the low probability of an event requiring this Function during the extended period. The 8 hour Completion Time also provides sufficient time to reach MODE 2 without challenging plant systems. Therefore, this change will not involve a significant reduction in a margin of safety.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L6 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?

   This change proposes to delete the reliability group categorization column for the instruments in CTS Tables 4.1-1 and 4.1-2. The design information contained in this column is not assumed to be an initiator of any design basis accident; therefore, the probability of an accident is not increased by this change. The consequences of an accident occurring without this information in the Technical Specifications are the same as the consequences without this information. 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?

   This proposed change will not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated or maintained. Therefore, this 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?

   This change proposes to delete the reliability group categorization column for the instruments in CTS Tables 4.1-1 and 4.1-2. This information was used in the original issuance of the Technical Specifications to determine functional testing frequency. The Surveillance Frequencies specified in the proposed Specifications will ensure that the RPS instrumentation remains Operable. As a result, the change does not affect the current analysis assumptions. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L7 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?

   This change proposes to delete the listed requirements for the functional testing and calibration of specific instruments in CTS Tables 4.1-1 and 4.1-2, respectively, because the proposed definitions for Channel Functional Test and Channel Calibration provide the necessary guidance. The proposed change does not increase the probability of an accident because the proposed Surveillance Requirements still ensure that the instruments are adequately tested. The proposed change provides assurance that the associated RPS Functions are tested consistent with the analysis assumptions. As a result, the consequences of an accident are not affected by this change. This change will not alter assumptions relative to the mitigation of an accident or transient event. 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?

   This change will not physically alter the plant (no new or different types of equipment will be installed). The changes in methods governing normal plant operation and testing are consistent with the current safety analysis assumptions. Therefore, this 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?

   This change proposes to delete specific testing information in CTS Tables 4.1-1 and 4.1-2 which is adequately addressed in the proposed definitions for Channel Functional Testing and Channel Calibration. The proposed change still provides the necessary control of testing to ensure Operability of the RPS instrumentation.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L7 CHANGE

3. (continued)

The safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L8 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 requirement to test the automatic scram contactors following maintenance is not assumed in the initiation of any analyzed event. This requirement was specified in the Current Technical Specifications to ensure the Operability of the automatic scram contactors was positively verified following maintenance. This explicit requirement is considered unnecessary because SR 3.0.1 requires the appropriate SRs to be performed to demonstrate Operability following restoration of a component that could cause the SR to be failed. In this case, SR 3.0.1 would require SR 3.3.1.1.4 to be performed, which would verify that the automatic scram contactors function properly. As a result, the accident consequences are unaffected by 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 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 proposed deletion of the explicit requirement to test the automatic scram contactors following maintenance is considered administrative because SR 3.0.1 requires the appropriate SRs to be performed to demonstrate Operability following restoration of a component that could cause the SR to be failed. In this case, SR 3.0.1 would require SR 3.3.1.1.4 to be performed. As a result, the existing requirement to test the automatic scram contactors following maintenance is maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

AMD 265

L9 CHANGE

Not Used.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L10 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?

   This change proposes to add a Note to the 7 day Channel Functional Test Surveillance Requirement for the IRM High Flux, IRM Inop and APRM Neutron Flux-High (Startup) Functions. The Note will allow the plant to enter MODE 2 from MODE 1 without performing the 7 day Channel Functional Test. The Surveillance, however, must be performed within 12 hours after entering MODE 2. The proposed change does not increase the probability of an accident. The Surveillance for the Channel Functional Test is not assumed to be an initiator of any analyzed event. The proposed change still provides assurance the associated RPS Functions are maintained consistent with analysis assumptions. The Notes allow time once in MODE 2 to perform the Surveillance because the associated IRM and APRM Functions cannot be performed in MODE 1 without utilizing jumpers or lifted leads. The 12 hour time limit is based on operating experience and the necessity to provide a reasonable time in which to complete the Surveillance Requirement. The proposed change provides confirmation of the Operability of the associated RPS Functions at the earliest opportunity when these Functions are required. In addition, the most common outcome of the performance of a Surveillance is the successful demonstration that the acceptance criteria are satisfied. As a result, the consequences of an accident are not affected by this change. This change will not alter assumptions relative to the mitigation of an accident or transient event. 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 create the possibility of an accident. This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L10 CHANGE

2. (continued)

normal plant operation are consistent with the current safety analysis assumptions. Therefore, this 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?

This change proposes to add a Note to the 7 day Channel Functional Test Surveillance Requirement for IRM High Flux, IRM Inop and APRM Neutron Flux-High (Startup) Functions. The Note will allow the plant to enter MODE 2 from MODE 1 without performing the 7 day Channel Functional Test. The Surveillance, however, must be performed within 12 hours after entering MODE 2. The margin of safety is not significantly reduced because the proposed change to the Surveillance Frequency will continue to provide the necessary assurance of Operability of the associated RPS Functions at the earliest opportunity. These changes effectively extend the initial performance of the Surveillance Requirement by 12 hours. This is considered acceptable since the most common outcome of the performance of a Surveillance is the successful demonstration that the acceptance criteria are satisfied. In addition, these changes provide the benefit of allowing the Surveillance to be postponed until plant conditions exist where the Surveillance can be performed without utilizing jumpers or lifted leads. The safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L11 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 delete the Instrument I.D. numbers for the RPS Instrumentation. The RPS Instrumentation is not considered as an initiator of any previously evaluated accident. The proposed change will not impact the ability of the RPS Instrumentation to perform its intended function. Therefore, the proposed change will not increase the probability of any accident previously evaluated. Additionally, while the RPS Instrumentation is assumed to mitigate accidents, this change does not affect the capability of the RPS Instrumentation to initiate a reactor scram when needed. Therefore, the proposed change will not increase the consequences of any 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 physical modification to the plant. The RPS Instrumentation provides signals to initiate a reactor scram. However, under the proposed change, Operability of the RPS Instrumentation is not impacted. Therefore, it 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 proposed change would delete the Instrument I.D. numbers for the RPS Instrumentation. However, these details are not necessary to ensure the RPS Instrumentation is maintained Operable. The requirements of ITS 3.1.1 (which describes the instrumentation) and associated Surveillance Requirements are adequate to ensure the required instrumentation is maintained Operable. The proposed change will not impact the ability of the RPS Instrumentation to perform its intended function. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L12 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 effectively extends the initial Surveillance Frequency until 12 hours after Thermal Power is ≥ 25% RTP. This allows time after the appropriate conditions are established to perform the Surveillance. The Surveillance is not required to be performed below 25% RTP because it is difficult to accurately determine core Thermal Power from a heat balance at these low power levels. In addition, at low power levels, a high degree of accuracy between the APRM indication and actual core Thermal Power is unnecessary due to the large inherent margin to the power distribution (thermal) limits at these power levels. This change does not increase the probability of an accident because the APRM RPS instrumentation is not assumed in the initiation of any analyzed event. The role of this instrumentation is in mitigating and, thereby, limiting the consequences of analyzed events. The proposed change still provides assurance the associated RPS Functions are maintained consistent with the analysis assumptions. The SR will still be performed at the earliest opportunity when these Functions are required. In addition, the most common outcome of the performance of a surveillance is the successful demonstration that the acceptance criteria are satisfied. As a result, the consequences of an accident are not affected by this change. This change will not alter assumptions relative to the mitigation of an accident or transient event. 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?

   This change will not alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L12 CHANGE

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

The margin of safety is not reduced by this change since the proposed change to the Surveillance Frequency provides the necessary assurance that the APRM instrumentation has been accurately calibrated at the earliest opportunity. This change extends the initial performance of the Surveillance Requirement to within 12 hours after reaching 25% RTP. This is considered acceptable since below 25% RTP a high degree of accuracy between the APRM indication and actual core Thermal Power is unnecessary due to the large inherent margin to the power distribution (thermal) limits at these power levels. In addition, this change provides the benefit of allowing the Surveillance to be postponed until appropriate plant conditions exist for performing the Surveillance accurately. The safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L13 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?

   This change relaxes the Surveillance Frequency for the performance of the APRM heat balance calibration from once per day to once per week. The proposed change does not affect the probability of an accident. The APRMs are not assumed to be an initiator of any analyzed event. The proposed change still provides assurance the APRMs are maintained consistent with analysis assumptions. The consequences of an accident are not affected by decreasing the frequency of the Surveillance to verify the APRM heat balance since the most common outcome of the performance of a surveillance is the successful demonstration that the acceptance criteria are satisfied. This change will not alter assumptions relative to the mitigation of an accident or transient event. 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?

   This change relaxes the Surveillance Frequency for performance of the APRM heat balance calibration from once per day to once per week. The proposed changes to the Frequency will not create the possibility of an accident. This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, this 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?

   This change relaxes the Surveillance Frequency for the performance of the APRM heat balance calibration from once per day to once per week. The increased Surveillance interval is acceptable since the once per

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L13 CHANGE

3. (continued)

week Frequency has been shown, based on industry operating experience, to be adequate for maintaining the APRMs consistent with the heat balance. Therefore, the margin of safety is not significantly reduced because the proposed changes to the Surveillance Frequency will continue to provide the necessary assurance that the APRM is being maintained within limits. Also, this change is considered acceptable since the most common outcome of the performance of a Surveillance is the successful demonstration that the acceptance criteria are satisfied. The safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L14 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 modifies the Trip Setting/Trip Level Setting (Allowable Value) in CTS 2.1.A.3 and CTS Table 3.1-1, Trip Function 15, Turbine Stop Valve Closure from ≤ 10% valve closure to ≤ 15% valve closure (ITS Table 3.3.1.1-1, Function 8) and the Trip Setting/Trip Level Setting (Allowable Value) in CTS 2.1.A.4 and CTS Table 3.1-1, Trip Function 14, Turbine Control Valve Fast Closure from > 500 psig and < 850 psig to ≥ 500 psig and ≤ 850 psig (ITS Table 3.3.1.1-1, Function 9). As discussed in Section 14.5.2.1 of the JAF UFSAR, the Turbine Control Valve Fast Closure instrumentation channels are actuated when a generator load rejection occurs, to avoid excessive turbine overspeed. As discussed in Section 14.5.2.2 of the JAF UFSAR, the Turbine Stop Valve Closure instrumentation channels are actuated whenever a turbine or reactor system malfunction occurs which may threaten turbine operation. Accordingly, these instrumentation channels are not an assumed initiator of any analyzed event as they respond to malfunctions in plant systems. In addition, existing operating margin between plant conditions and actual plant setpoints is not significantly reduced due to this change. As a result, the proposed changes will not result in unnecessary plant transients. The role of the instrumentation in ITS 3.3.1.1 is in the mitigating and thereby limiting the consequences of accidents. The Allowable Values and Trip Setpoints have been developed to ensure that the design and safety analysis limits will be satisfied. The methodology used for the development of the Allowable Values and Trip Setpoints ensures the affected instrumentation remains capable of mitigating design basis events as described in the safety analysis and that the results and consequences described in the safety analysis remain bounding. Additionally, the proposed change does not alter the plant's ability to detect and mitigate events. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

TECHNICAL CHANGE - LESS RESTRICTIVE (SPECIFIC)

L14 CHANGE

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 create the possibility of a new or different kind of accident from any previously evaluated. This is based on the fact that the method and manner of plant operation is unchanged. The use of the proposed Allowable Values and Trip Setpoints does not impact safe operation of the James A. FitzPatrick Nuclear Power Plant in that the safety analysis will be satisfied. The proposed Allowable Values and Trip Setpoints involve no system additions or physical modifications to systems at the plant. These Allowable Values and Trip setpoints were developed using a methodology to ensure the affected instrumentation remains capable of mitigating accidents and transients.

   Plant equipment will not be operated in a manner different from previous operation, except that setpoints will be changed. Since operational methods remain unchanged and the operating parameters have been evaluated to maintain the plant within existing design basis criteria, no different type of failure or accident is created.

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

   The proposed change does not involve a reduction in a margin of safety. The proposed changes have been developed using a methodology to ensure safety analysis limits are not exceeded. As such, this proposed change does not involve a significant reduction in the margin of safety.

F

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.3.1.1

Reactor Protection System (RPS) Instrumentation

MARKUP OF NUREG-1433, REVISION 1 SPECIFICATION

3.3 INSTRUMENTATION

[3.1.A] 3.3.1.1 Reactor Protection System (RPS) Instrumentation

[3.1.A] LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

[Table 3.1-1] [2.1.A] APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

-----NOTE-----
[A2] Separate Condition entry is allowed for each channel.

	CONDITION	REQUIRED ACTION	COMPLETION TIME
[T. 3.1-1 Note 1.a]	A. One or more required channels inoperable.	A.1 Place channel in trip. OR A.2 Place associated trip system in trip.	12 hours 12 hours
[T.3.1-1 note 1b] [Note 1.b.(2)]	B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. OR B.2 Place one trip system in trip.	6 hours 6 hours
[T.3.1-1 Note 1.b.(1)]	C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

(continued)

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME
[T3.1-1 nt 1.b]	D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
[T3.1-1 nt 3.C]	E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < $\boxed{30}$% RTP. 29 — DB1	4 hours
[T3.1-1 nt 3.G] [L5]	F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	~~6~~ hours 8 — CLB 11
[T3.1-1 nt 3.A]	G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
[L3]	H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

[4.1.A]

[3.1-1 Nt 2]

---NOTES---

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

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

(continued)

[Table 4.1-2]	SR 3.3.1.1.9 Perform CHANNEL CALIBRATION.	92 days

DB7

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.11	-------------------NOTES------------------- 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	24 [18] months CLB6
SR 3.3.1.1.13	-------------------NOTES------------------- 1. Neutron detectors are excluded. 2. For Function ~~1~~ a, PA3 not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL CALIBRATION.	DB8 24 [18] months
~~SR 3.3.1.1.14~~	~~Verify the APRM Flow Biased Simulated Thermal Power—High time constant is ≤ [7] seconds.~~	~~[18] months~~
SR 3.3.1.1.~~15~~ 14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 [18] months X1

(continued)

[M11]
[T. 4.1-1]
[4.1-2]
[M9]
DB2
[M4]
DB2
F

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
DB2 SR 3.3.1.1.~~16~~ 15	Verify Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, ~~Trip~~ EHC Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ ~~[30]~~ 29 % RTP. DB1	~~[18]~~ 24 months X2 PA1
DB2 SR 3.3.1.1.~~17~~ 16	-------------------NOTES------------------- 1. Neutron detectors are excluded. 2. ~~For Function 5~~ "n" equals ~~3~~ 2 channels for the purpose of determining the the STAGGERED TEST BASIS Frequency. -- Verify the RPS RESPONSE TIME is within limits.	CLB10 ~~[18]~~ 24 months on a STAGGERED TEST BASIS

[M13]

[4.1.A]
[MB]

RAI 3.3.1.1-6
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Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

PA2

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors PA3					
[2.1.A.1.a] [T.3.3-1(3)] [T.4.1-1(4)] [T.4.1-2(1)] a. Neutron Flux — High	2	[3]	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.4	≤ [120/125] divisions of full scale DB9
	5[(a)]	[3] CLB3	H	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ [120/125] divisions of full scale
[Table 3.3-1(4)] [T.4.1-1(5)] b. Inop	2	[3]	G	SR 3.3.1.1.7 SR 3.3.1.1.14	NA
	5[(a)]	[3]	H	SR 3.3.1.1.3 SR 3.3.2.2.14	NA
2. Average Power Range Monitors PA1					
[2.1.A.1.b] [T.3.3-1(5)] [T.4.1.2(4)] [T.4.1-1(9)] [T.4.1-2(2)] a. Neutron Flux — High, ~~Setdown~~ (Startups)	2	[2] CLB3	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.7 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.4	≤ [15]% RTP DB9
[T.4.1-2(3)] [T.4.1.2(4)] [2.1.A.1.c.(1)] [Table 3.3-1(6)] [T.4.1-1(8)] [T3.1-1 Notes 12 & 13] [4.1.A.8] b. ~~Flow Biased Simulated Thermal Power — High~~ Neutron Flux - High (Flow Biased) DB2	1	[2] DB3	F	SR 3.3.1.1.1 SR 3.3.1.1.2 ~~SR 3.3.1.1.8~~ SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.16	~~≤ [0.58 W + 62]% RTP and ≤ [115.5]% RTP[(b)]~~ As specified in the COLR DB9

(continued)

[T3.1-1 Note 7.D] [L1]

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

~~(b) [0.58 W + 62% - 0.58 ΔW]RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."~~ DB9

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
[T.4.1-1(6)] [T.3.3-1(7)] [2.1.A.1.c(2)] [4.1.A.7] [T.4.1-2(4)] 2. Average Power Range Monitors (continued)					
c. (Fixed) Neutron Flux — High (PA1)	1	[2] (DB3)	F (CLB3)	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 (7) SR 3.3.1.1.9 (8) SR 3.3.1.1.11 SR 3.3.1.1.15 (14) SR 3.3.1.1.17 (16) SR 3.3.1.1.4	≤ [120]% RTP (CLB2) (DB9)
d. Downscale (CLB8)	1	[2]	F	SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.15	≥ [3]% RTP
[T.4.1-1(7)] [T.3.3-1(8)] (d) e. Inop (CLB8) (PA1)	1,2	[2]	G	SR 3.3.1.1.8 (7) SR 3.3.1.1.9 (8) SR 3.3.1.1.15 (14)	NA
[2.2.1.A] [MA6] [4.1.A.1] [T.3.3-1(9)] [T.4.1-1(10)] [T.4.1-2(5)] 3. Reactor Vessel Steam Dome Pressure — High (PA1)	1,2	[2]	G	SR 3.3.1.1.1 SR 3.3.1.1.9 (8) [SR 3.3.1.1.10] SR 3.3.1.1.13 SR 3.3.1.1.15 (14) SR 3.3.1.1.17 (16)	≤ [1054] 1079 psig (DB9)
[2.1.A.2] [4.1.A.3] [T.4.1-2(7)] [T.3.3-1(11)] [T.4.1-1(12)] 4. Reactor Vessel Water Level — Low, (Level 3) (1A1)	1,2	[2]	G	SR 3.3.1.1.1 SR 3.3.1.1.9 (8) [SR 3.3.1.1.10] SR 3.3.1.1.13 (14) SR 3.3.1.1.15 SR 3.3.1.1.17 (16)	≥ [10] 177 inches (DB9)
[T.4.1-1(15)] [2.1.A.5] [4.1.A.4] [T.3.3-1(13)] [T.4.1-2(10)] 5. Main Steam Isolation Valve — Closure	1	[8] 8	F	SR 3.3.1.1.9 (8) SR 3.3.1.1.13 (14) SR 3.3.1.1.15 (16) SR 3.3.1.1.17	≤ [10]% closed 15 (DB9)
[4.1.A.2] [T.3.3-1(10)] [T.4.1-1(6)] [T.4.1-2(6)] 6. Drywell Pressure — High	1,2	[2] (DB3)	G	SR 3.3.1.1.1 SR 3.3.1.1.9 (8) [SR 3.3.1.1.10] SR 3.3.1.1.13 SR 3.3.1.1.15 (14) SR 3.3.1.1.16 (CLB9)	≤ [1.92] 2.7 psig (DB9)

(continued)

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Instrument Volume Water Level – High					
[T. 3.3-1 (12)] [T.4.1-1 (14)] [T.4.1-2 (9)] a. ~~Resistance Temperature Detector~~ Differential Pressure Transmitter / trip unit	1,2	~~[2]~~	G	SR 3.3.1.1.1 SR 3.3.1.1.9 ~~SR 3.3.1.1.10~~ ~~SR 3.3.1.1.13~~ SR 3.3.1.1.~~15~~ 14 SR 3.3.1.1.4	≤ ~~[57.15]~~ 34.5 gallons
	5[(a)]	~~[2]~~	H	SR 3.3.1.1.1 SR 3.3.1.1.9 ~~SR 3.3.1.1.10~~ ~~SR 3.3.1.1.13~~ SR 3.3.1.1.~~15~~ 14 SR 3.3.1.1.4	≤ ~~[57.15]~~ 34.5 gallons
[T. 4.1-2 (8)] [T. 3.3-1 (12)] [T. 4.1-1 (13)] b. ~~[Float]~~ Level Switch	1,2	~~[2]~~	G	SR 3.3.1.1.~~9~~ 8 SR 3.3.1.1.13 SR 3.3.1.1.~~15~~ 14 SR 3.3.1.1.4	≤ ~~[57.15]~~ 34.5 gallons
	5[(a)]	~~[2]~~	H	SR 3.3.1.1.~~9~~ 8 SR 3.3.1.1.13 SR 3.3.1.1.~~15~~ 14 SR 3.3.1.1.4	≤ ~~[57.15]~~ 34.5 gallons
[T. 4.1-1 (17)+(18)] [T. 4.1-2 (11)+(13)] [T. 3.3-1 (15)] [2.1.A.3] [V.1.A.5] [L14] 8. Turbine Stop Valve – Closure	≥ ~~[30]~~ 29% RTP	~~[4]~~	E	SR 3.3.1.1.9 8 ~~SR 3.3.1.1.10~~ SR 3.3.1.1.13 SR 3.3.1.1.~~15~~ 14 SR 3.3.1.1.~~16~~ 15 SR 3.3.1.1.~~17~~ 16 SR 3.3.1.1.1	≤ ~~[10]~~ 15% closed
[T.4.1-1 (16)+(17)] [T.4.1-2 (11)+12] [T. 3.3-1 (14)] [2.1.A.4] [V.1.A.6] [L14] 9. Turbine Control Valve Fast Closure, ~~[Trip]~~ EHC Oil Pressure – Low	≥ ~~[30]~~ 29% RTP	~~[2]~~	E	SR 3.3.1.1.~~9~~ 8 ~~SR 3.3.1.1.10~~ SR 3.3.1.1.13 SR 3.3.1.1.~~15~~ 14 SR 3.3.1.1.~~16~~ 15 SR 3.3.1.1.~~17~~ 16 SR 3.3.1.1.1	≥ ~~[600]~~ 500 psig and ≤ 850 psig
[T. 3.3-1 (1)] [T. 4.1-1 (1)] 10. Reactor Mode Switch – Shutdown Position	1,2	[2] 1	G	SR 3.3.1.1.12 SR 3.3.1.1.~~15~~ 14	NA
	5[(a)]	[2] 1	H	SR 3.3.1.1.12 SR 3.3.1.1.~~15~~ 14	NA
[T. 3.3-1 (2)] [T. 4.1-1 (2)] 11. Manual Scram	1,2	[2] 1	G	SR 3.3.1.1.~~9~~ 8 SR 3.3.1.1.~~15~~ 14	NA
	5[(a)]	[2] 1	H	SR 3.3.1.1.~~9~~ 8 SR 3.3.1.1.~~15~~ 14	NA

PA1 PA2 DB3 DB4 CLB3 DB5 DB9 DB1 DB6 1E 1A

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

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.3.1.1

Reactor Protection System (RPS) Instrumentation

JUSTIFICATION FOR DIFFERENCES (JFDs)
FROM NUREG-1433, REVISION 1

RETENTION OF EXISTING REQUIREMENT (CLB)

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

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

CTS 4.1.2 "Flow Biased Signal" requires an "internal power and flow test with standard pressure source" calibration on a "refueling interval," which has been translated into ITS SR 3.3.1.1.11. This calibration of the flow signal is at a frequency that is consistent with the current licensing basis. The Functional Test of the APRMs (ITS SR 3.3.1.1.6) is consistent with CTS Table 4.1-1, which ensures the APRM circuitry responds appropriately to this calibrated flow signal. As such, the proposed ITS adequately translates the current licensing basis for testing the APRM Flow Biased Function without adopting the ISTS SR 3.3.1.1.3.

RAI 3.3.1.1-8

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

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

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

CLB6 The brackets have been removed from the CHANNEL FUNCTIONAL TEST Frequency in ITS SR 3.3.1.1.12 and extended from 18 months to 24 months consistent with the Channel Functional Test frequencies of CTS Table 4.1-1. The Frequency is consistent with the JAFNPP fuel cycle.

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CLB7 Not Used.

RETENTION OF EXISTING REQUIREMENT (CLB)

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

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

CLB10 Note 3 of ITS SR 3.3.1.1.16 has been changed to ensure that all channels are tested within two surveillance intervals consistent with the current licensing basis. In addition, the bracketed SR Frequency has been changed from 18 to 24 months consistent with the current Frequency in CTS 4.1.A.

CLB11 The Completion Time associated with ITS 3.3.1.1 Required Action F.1 (Be in MODE 2) has been extended from 6 hours to 8 hours. This proposed Completion Time is consistent with CTS Table 3.1-1 Note 3.B. The CTS Actions for the Main Steam Line Closure Function has been modified as described in L5. The proposed time of 8 hours is considered reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems.

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

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

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

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

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

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

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

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

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

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

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

DB5 For Function 7.a, ITS SR 3.3.1.1.10, the calibration of the trip unit, and ITS SR 3.3.1.1.13, the CHANNEL CALIBRATION test every 18 months, has been deleted since this Function is calibrated in accordance with ITS SR 3.3.1.1.9 every 92 days. Since this calibration includes the entire channel this specific requirement to calibrate the trip units, is not necessary. The 92 day CHANNEL CALIBRATION Frequency is consistent with the methodology for the setpoint calculation of this Function.

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

DB7 ITS SR 3.3.1.1.9 has been added to perform a CHANNEL CALIBRATION every 92 days for Function 7.a (Scram Discharge Instrument Volume Water Level – High, Differential Pressure Transmitter/Trip Unit) consistent with CTS Table 4.1-2. The Frequency is consistent with the setpoint calculation methodology for this Function.

DB8 The brackets have been removed from the Surveillance Frequency in ITS SR 3.3.1.1.13 (CHANNEL CALIBRATION) and extended from 18 months to 24 months consistent with the frequencies in CTS Table 4.1-2 and as justified in M9 for the IRM High Flux channels. The Frequency is consistent with the setpoint calculation methodology for the associated Functions.

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

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

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

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

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 The brackets have been removed from the Frequency in ITS SR 3.3.1.1.14 (the LOGIC SYSTEM FUNCTIONAL TEST) and the 18 month surveillance extended to 24 months as justified in M4. This Frequency is consistent with the JAFNPP fuel cycle.

X2 The brackets have been removed from the Frequency in ITS SR 3.3.1.1.15 (the verification bypass feature) and the 18 month surveillance extended to 24 months as justified in M13. This Frequency is consistent with the JAFNPP fuel cycle.