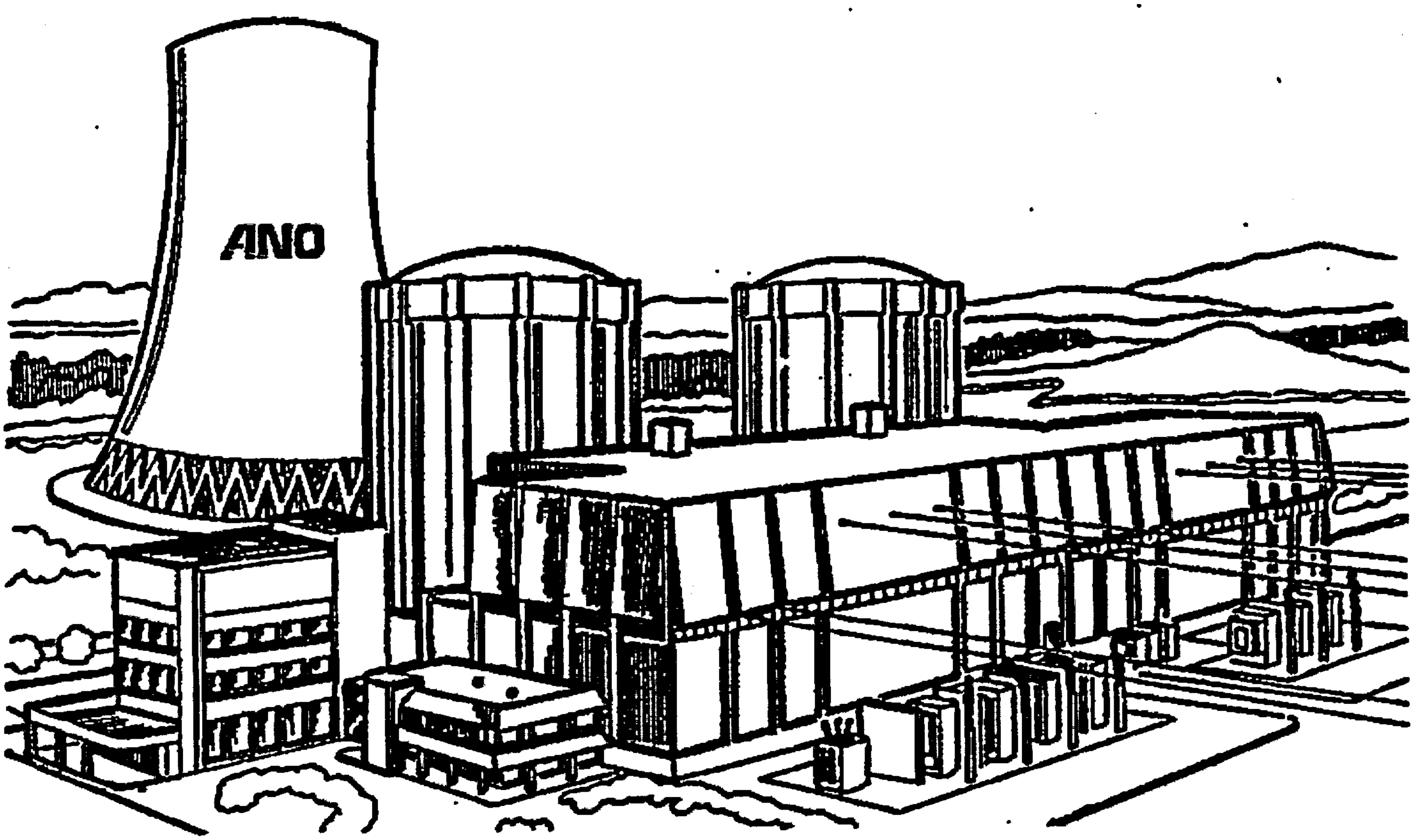
ARKANSAS NUCLEAR ONE - UNIT 1

IMPROVED TECHNICAL SPECIFICATIONS SUBMITTAL



VOLUME 1 OF 7

Entergy

January 28, 2000

SYNOPSIS OF THE PROPOSED LICENSE AMENDMENT REQUEST

The submittal for the conversion to Improved Technical Specifications (ITS) consists of 7 volumes and related attachments to the transmittal letter. The 7 volumes consist of the application of NRC Selection Criteria (Split Report) and ITS Section packages. Below is a brief description of the contents of the Split Report and each of the Section packages, as well as a brief explanation of how the material was prepared and the designations utilized.

APPLICATION OF NRC SELECTION CRITERIA

The Selection Criteria provides a discussion of how the 10 CFR 50.36(c)(2)(ii) criteria were evaluated with respect to the Current Technical Specification (CTS) requirements. This evaluation was performed only to determine those CTS requirements that are required to be retained in the proposed ITS. For those CTS requirements that do not meet any of the NRC selection criteria and are not retained in the proposed ITS, an evaluation of the CTS requirement against the criteria is provided in Appendix A.

SECTION PACKAGES FOR SECTIONS 1.0 THROUGH 5.0 (18 SECTIONS)

Each of the Section packages corresponds to a Section of the proposed ITS. ITS Section 3.3, "Instrumentation," has been divided into four sub-Sections based on the instrumentation systems (Reactor Protection System, Engineered Safeguards Systems, Emergency Feedwater Indication and Controls, and Miscellaneous Instrumentation). ITS Section 3.4, "Reactor Coolant Systems," has been divided into two sub-Sections based on the systems (Reactor Coolant System and Reactor Coolant System Auxiliaries). Each Section, and sub-Section contains the required information to review the ITS Section, and is organized as described below:

TAB <u>ITS</u>

Contains the proposed ITS Limiting Conditions for Operation (LCOs).

TAB <u>ITS Bases</u>

Contains the proposed ITS Bases

TAB <u>CTS Markup</u>

Contains annotated copies of the CTS pages which show the disposition of existing requirements into the proposed ITS. The pages are arranged in ITS order. The upper right hand corner of the CTS page is annotated with the ITS Specification number to which the CTS page applies. Items on the CTS page that are addressed in other proposed ITS Sections (or Specifications within the Section) are annotated with the appropriate location.

Where a proposed ITS requirement differs from a CTS requirement, individual details of the CTS revision are annotated with alpha-numeric designators which relate to the appropriate Discussion of Change (DOC). The DOC provides a concise justification for the change. The DOCs are located directly preceding the CTS Markup in each Section or sub-Section. The alpha-numeric designators also relate to the evaluations supporting a finding of No Significant Hazard Consideration (NSHC).

The CTS pages in the Section packages reflect License Amendments issued as of the date of the submittal letter, and License Amendment Requests described in Attachment 2 to the submittal letter.

The DOCs are numbered sequentially within each letter category for each ITS Section or sub-Section. The proposed changes for each CTS requirement are separated into the following categories:

Designator	Category
A	ADMINISTRATIVE - changes to the CTS that result in no additional or reduced restrictions or flexibility. These changes are supported in aggregate by a single NSHC.
M	TECHNICAL CHANGES - MORE RESTRICTIVE - changes to the CTS that result in added restrictions or reduced flexibility. These changes are supported in aggregate by a single NSHC.
L	TECHNICAL CHANGES - LESS RESTRICTIVE - changes to the CTS that result in reduced restrictions or added flexibility. Each corresponding evaluation is supported by a corresponding evaluation supporting a finding of NSHC.
LA	TECHNICAL CHANGES - REMOVAL OF DETAIL - changes to the CTS that eliminate detail and relocate the detail to a licensee controlled document. Typically, this involves details of system design and function, or procedural detail on methods of conducting a surveillance. These changes are supported in aggregate by a single NSHC.

R RELOCATED SPECIFICATIONS - changes to the CTS that encompass the requirements that do not meet the selection criteria of 10 CFR 50.36(c)(2)(ii). These changes are supported in aggregate by a single NSHC.

The CTS Bases pages are replaced in their entirety. A single DOC justifies the replacement.

TAB NSHC

Contains evaluations required by 10 CFR 50.91(a) supporting a finding of No Significant Hazard Consideration (NSHC). Generic evaluations for a finding of NSHC have been written for each category of changes except Category "L." The evaluations supporting a finding of NSHC are ordered as follows: A, M, LA, R, and L. Each evaluation is annotated to correspond to the DOC discussed in the NSHC. The generic NSHC evaluations for Category A, M, and R changes are located in the Split Report section.

TAB NUREG Markup

Contains annotated copies of the applicable NUREG-1430, Revision 1, LCOs which show how the proposed ITS LCO differs from the NUREG LCO. Where a proposed ITS LCO differs from the NUREG LCO, individual details of the change are annotated with numeric designators which relate to the appropriate Discussion of Difference (DOD). The DOD provides a concise justification for the change. The LCO DODs are located directly preceding the associated markup for each Section or sub-Section.

TAB Bases Markup

Contains annotated copies of the applicable NUREG-1430, Revision 1, Bases which show how the proposed ITS Bases differ from the NUREG Bases. Where a proposed ITS Bases requirement differs from the NUREG Bases, individual details of the change are annotated with numeric designators which relate to the appropriate DOD. The DOD provides a justification for the change. The DODs are located directly preceding the associated markup of the NUREG Limiting Conditions for Operation for each Section or sub-Section.

CTS MARKUP IN CTS ORDER

The individual section CTS Markups in the Section packages were prepared in the order of the ITS. This volume provides an entire markup of the CTS in CTS order to demonstrate that all CTS requirements are accounted for. In many instances, the same CTS page is used in several ITS Sections. As a result, in this volume, the same CTS page will appear with the annotations associated with the ITS Section or sub-Section in which it appears.

TREATMENT OF INSTRUMENT UNCERTAINTY

During the development of the ANO-1 ITS, parameter values were reviewed to determine whether the CTS value contained allowances for instrument uncertainty. This review incorporated the results of the ANO-1 Plant Parameter Uncertainty Project (PPUP). The PPUP is an ANO design basis initiative based on Entergy Operations' graded approach methodology for determining and documenting uncertainties in the measurement of plant parameters. The purpose of PPUP is to evaluate all parameter values contained in the technical specifications and determine how instrument uncertainty should be characterized. The parameter values associated with the Reactor Protection System, Engineered Safeguards Actuation System, and Emergency Feedwater Initiation and Control System trip setpoints are characterized as "Allowable Values" containing allowances for instrument uncertainties. The actual trip setpoints entered into the bistables for these trip parameters are conservative with respect to the Allowable Values, and are controlled in the implementing procedures. Other parameter values contained in the ITS have been stated without allowances for instrument uncertainty. In other words, these values reflect the value assumed in the safety analyses (where appropriate) with no allowance for instrument uncertainty. Unless otherwise noted, ITS parameter values do not contain an allowance for instrument uncertainty in the measurement of the parameter. These changes are considered to be administrative in nature. Entergy Operations proposes to control any required instrument uncertainties in the implementing procedures.

ENVIRONMENTAL IMPACT EVALUATION

10 CFR 51.22(c) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released off-site, or (3) result in a significant increase in individual or cumulative occupational radiation exposure. Entergy Operations, Inc. has reviewed this license amendment and has determined that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the proposed license amendment. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described previously in the evaluation.

2. As discussed in the significant hazards evaluation, this change does not result in a significant change or significant increase in the radiological doses for any Design

Basis Accident. The proposed license amendment does not result in a significant change in the types or a significant increase in the amounts of any effluents that may be released off-site.

3. The proposed license amendment does not result in a significant increase to the individual or cumulative occupational radiation exposure because this change has been shown to retain controls necessary to limit exposures.

EXISTING ANO-1 LICENSE AMENDMENT REQUESTS INCORPORATED INTO THE ITS

DATE	LETTER	DESCRIPTION	AFFECTED ITS SPEC	CTS PAGE(S)
August 6, 1998	1CAN089801	Revision of the Sodium Hydroxide Tank limits	3.6.7	37 and 39
July 14, 1999	0CAN079901	Deletion of post accident sampling system requirements	N/A	126
August 18, 1999	1CAN089903	Revision to Engineered Safeguards Actuation System low reactor coolant system pressure setpoint	3.3.5	49 and 50
September 17, 1999	0CAN099901	Revision of the curie limits for radioactive gas storage tanks	5.5.12	66w
November 23, 1999	0CAN119906	Revision of charcoal filter testing requirements per GL 99-02	5.5.11	66c, 66d, 66g, 66h, 109, 109a, and 110I
January 27, 2000	0CAN010004	Revision of Condensate Storage Tank volume	3.7.6	40 and 41a

DISPOSITION OF GENERIC CHANGES TO NUREG-1430

TSTF NUMBER	NRC STATUS	ITS SECTION	STATUS[1]	NOTES	DISCUSSION
TSTF-001, R 01	Rejected	3.0	NI	Not incorporated due to rejection	
TSTF-002, R 01	Approved	3.8	INC		3.8 DOD-20
TSTF-003, R 01	Rejected	1.0 3.4A 3.4B 3.7	NI NI NI NI	Not incorporated due to rejection	
TSTF-004, R 01	Rejected	1.0	NI	Not incorporated due to rejection	
TSTF-005, R 01	Approved	2.0	IWC	Incorporated with the exception that NUREG-1430 2.2.5 has been retained as a site preference.	2.0 DOD-03
TSTF-006, R 01	Approved	3.0	INC		3.0 DOD-01
TSTF-007, R 01	Withdrawn	3.0	NI	Not incorporated due to withdrawal	
TSTF-008, R 02	Approved	3.0 3.8	INC INC		3.0 DOD-05 3..8 DOD-08
TSTF-009, R 01	Approved	3.1 3.2 3.3A	INC INC INC		3.1 DOD-01 3.2 DOD-02 3.3a DOD-28
TSTF-016, R 02	Pending	3.8	NI	Not incorporated due to not being approved	
TSTF-017, R 02	Approved	3.6	IWC	ANO revised to maintain consistency with fuel cycle length.	3.6 DOD-08

TSTF NUMBER	NRC STATUS	ITS SECTION	STATUS[1]	NOTES	DISCUSSION
TSTF-019, R 01	Approved	1.0 3.3A 3.3D	INC INC INC		1.0 DOD-12 3.3a DOD-29 3.3d DOD-30
TSTF-020, R 00	Approved	3.9	INC		3.9 DOD-21
TSTF-021, R 01	Rejected	3.9	NI	Not incorporated due to rejection	
TSTF-022, R 00	Rejected	3.9	NI	Not incorporated due to rejection	
TSTF-026, R 00	Approved	3.4A	NI	Not incorporated due to retention of current license basis	
TSTF-027, R 03	Approved	3.4A	IWC	Revised to reflect unit specific design	3.4a DOD-06
TSTF-028, R 00	Approved	3.4B	IWC	Required Actions for ITS 3.4.12 (NUREG-1430 LCO 3.4.16) have been revised to reflect the ANO-1 current license basis, in addition to the incorporation of TSTF-028	3.4b DOD-19
TSTF-030, R 03	Approved	3.6	IWC	ANO-1 is not licensed as an SRP plant. Therefore, Reference 6 has not been incorporated	3.6 DOD-06
TSTF-036, R 04	Approved	3.7 3.8	NI NI	Not incorporated due to retention of current license basis Not incorporated due to not including NUREG-1430 LCO 3.8.2, 3.8.5, 3.8.8, & 3.8.10 in the ANO-1 ITS	
TSTF-037, R 02	Approved	3.3D 3.8 5.0	NI NI INC	Not incorporated in NUREG-1430 3.8.1 and 3.3.17 due to retention of current license basis	5.0 DOD-19
TSTF-038, R 00	Approved	3.8	NI	Not incorporated due to not including NUREG-1430 SR 3.8.4.3 in the ANO-1 ITS	

TSTF NUMBER	NRC STATUS	ITS SECTION	STATUS[1]	NOTES	DISCUSSION
TSTF-039, R 01	Withdrawn	1.0	NI	Not incorporated due to withdrawal	
TSTF-040, R 00	Approved	1.0	IWC	The reference to seal "injection" has not been incorporated since leakage is associated with escape of fluids from systems or boundaries. This is consistent with the current license basis	1.0 DOD-13 1.0 DOD-14
TSTF-041, R 00	Pending	3.0	NI	Not incorporated due to not being approved	
TSTF-042, R 00	Approved	3.0	INC		3.0 DOD-04
TSTF-043, R 00	Approved	3.0	INC		3.0 DOD-09
TSTF-044, R 00	Rejected	3.6	NI	Not incorporated due to rejection	
TSTF-045, R 02	Approved	3.6	INC		3.6 DOD-04
TSTF-046, R 01	Approved	3.6	INC		3.6 DOD-05
TSTF-051, R 02	Approved	3.3D 3.6 3.7 3.8 3.9	NI NI NI NI NI	Approved too late in ITS development cycle for inclusion	
TSTF-052, R 02	Pending	1.0 3.0 3.6 5.0	NI NI NI NI	Not incorporated due to not being approved	
TSTF-054, R 01	Approved	3.4B	INC		3.4b DOD-26

TSTF NUMBER	NRC STATUS	ITS SECTION	STATUS[1]	NOTES	DISCUSSION
TSTF-056, R 00	Approved	3.4B	NI	Not incorporated based on not including NUREG-1430 LCO 3.4.11 in the ANO-1 ITS	
TSTF-057, R 00	Approved	3.4B	INC		3.4b DOD-18
TSTF-060, R 00	Approved	3.4B	INC		3.4b DOD-28
TSTF-061, R 00	Approved	3.4B	INC		3.4b DOD-27
TSTF-063, R 00	Approved	3.4A	NI	Superceded by changes made per TSTF-263 and incorporated	
TSTF-064, R 00	Withdrawn	1.0	NI	Not incorporated due to withdrawal	
TSTF-065, R 01	Approved	2.0 5.0	NI IWC	Revised to reflect current ANO-1 license basis	5.0 DOD-01 5.0 DOD-33
TSTF-068, R 02	Approved	3.9	NI	Not incorporated due to incorporation of current license basis.	
TSTF-070, R 01	Approved	3.7	IWC	Incorporated with editorial changes to reflect ANO-1 naming conventions and to enhance readability of Bases inserts.	3.7 DOD-30
TSTF-071, R 02	Approved	3.0	NI	Not incorporated because site reviewers felt that the level of detail contained in the Bases without the change was appropriate.	
TSTF-086, R 00	Rejected	5.0	NI	Not incorporated due to rejection	
TSTF-088, R 00	Withdrawn	1.0	NI	Not incorporated due to withdrawal	
TSTF-090, R 01	Approved	3.5	IWC	TSTF-090 was revised to retain the SR 3.5.3.1 Note in the Surveillance Requirement as well as echoing it in the LCO to avoid confusion with respect to the applicable SRs.	3.5 DOD-12

TSTF NUMBER	NRC STATUS	ITS SECTION	STATUS[1]	NOTES	DISCUSSION
TSTF-091, R 01	Rejected	3.3D	NI	Not incorporated due to rejection	
TSTF-092, R 01	Modify	3.9	IWC	TSTF-092 insert was revised to incorporate current license basis	3.9 DOD-04
TSTF-096, R 01	Approved	3.9	INC		3.9 DOD-05
TSTF-100, R 00	Approved	3.7	NI	Not incorporated due to not including NUREG-1430 LCO 3.7.4 in the ANO-1 ITS	
TSTF-101, R 00	Approved	3.7	INC		3.7 DOD-13
TSTF-102, R 00	Rejected	3.7	NI	Not incorporated due to rejection	
TSTF-104, R 00	Approved	3.0	INC		3.0 DOD-11
TSTF-105, R 01	Rejected	3.4A	NI	Not incorporated due to rejection	
TSTF-106, R 01	Approved	5.0	INC		5.0 DOD-38
TSTF-107, R 04	Approved	3.1	NI	Changes consistent with the current license basis were incorporated in lieu of TSTF-107	
TSTF-110, R 02	Approved	3.1 3.2	INC INC		3.1 DOD-12 3.2 DOD-03 3.2 DOD-08 3.2 DOD-22
TSTF-113, R 04	Rejected	3.4B	NI	Not incorporated due to rejection	
TSTF-115, R 00	Withdrawn	3.8 5.0	NI NI	Not incorporated due to withdrawal	

TSTF NUMBER	NRC STATUS	ITS SECTION	STATUS(1)	NOTES	DISCUSSION
TSTF-116, R 02	Approved	3.4B	IWC	In addition to the incorporation of TSTF-116, additional clarification has been added that steady state operation is at or near normal operating pressure as a plant specific clarification based on operating history	3.4b DOD-11 3.4b DOD-17
TSTF-118, R 00	Approved	5.0	INC		5.0 DOD-16 5.0 DOD-35
TSTF-119, R 00	Rejected	5.0	NI	Not incorporated due to rejection	
TSTF-120, R 00	Rejected	5.0	NI	Not incorporated due to rejection	
TSTF-121, R 00	Withdrawn	5.0	NI	Not incorporated due to withdrawal	
TSTF-122, R 00	Approved	3.0	INC		3.0 DOD-07
TSTF-123, R 01	Approved	4.0	INC		4.0 DOD-02
TSTF-124, R 00	Approved	1.0	INC		1.0 DOD-08
TSTF-125, R 01	Approved	1.0 3.2	INC INC		1.0 DOD-07 3.2 DOD-13
TSTF-126, R 00	Approved	2.0	INC		2.0 DOD-02
TSTF-137, R 00	Approved	3.4B	NI	Not incorporated due to the deletion of the LCO 3.0.4 exception note for incorporation of current license basis required actions	
TSTF-138, R 00	Rejected	3.4B	NI	Not incorporated due to rejection	
TSTF-139, R 01	Approved	3.7	NI	Not incorporated due to not including NUREG-1430 LCO 3.7.14 in the ANO-1 ITS	

TSTF NUMBER	NRC STATUS	ITS SECTION	STATUS(1)	NOTES	DISCUSSION
TSTF-140, R 00	Approved	3.7	IWC	Reference to Final Policy Statement replaced with reference to 10CFR50.36	3.7 DOD-43
TSTF-141, R 01	Rejected	3.1	NI	Not incorporated due to rejection	
TSTF-142, R 00	Approved	3.1	INC		3.1 DOD-29
TSTF-143, R 00	Approved	3.1	INC		3.1 DOD-03
TSTF-145, R 01	Withdrawn	3.6	NI	Not incorporated due to withdrawal	
TSTF-152, R 00	Approved	5.0	INC		5.0 DOD-26
TSTF-153, R 00	Approved	3.4A 3.9	IWC NI	Revised Note proposed by TSTF-153 to clarify that pumps "may be removed from operation" in ITS Section 3.4. TSTF-153 was not incorporated in ITS Section 3.9 due to retention of current license basis.	3.4a DOD-05
TSTF-154, R 02	Approved	3.1	IWC	Reference to Final Policy Statement replaced with reference to 10CFR50.36	3.1 DOD-13
TSTF-155, R 01	Pending	3.5	NI	Not incorporated due to not being approved	
TSTF-156, R 01	Approved	3.1	INC		3.1 DOD-25
TSTF-157, R 01	Withdrawn	3.7	NI	Not incorporated due to withdrawal	
TSTF-158, R 01	Approved	3.1	INC		3.1 DOD-40
TSTF-159, R 01	Approved	3.1	INC		3.1 DOD-07

TSTF NUMBER	NRC STATUS	ITS SECTION	STATUS[1]	NOTES	DISCUSSION
TSTF-160, R 01	Approved	3.1 3.2	INC INC		3.1 DOD-20 3.1 DOD-33 3.2 DOD-01 3.2 DOD-06 3.2 DOD-23
TSTF-163, R 02	Approved	3.8	NI	Not incorporated due to not including NUREG-1430 SRs 3.8.1.7, 3.8.1.12, 3.8.1.15, and 3.8.1.20 in the ANO-1 ITS.	
TSTF-165, R 00	Approved	3.0	INC		3.0 DOD-12
TSTF-166, R 00	Approved	3.0	INC		3.0 DOD-06
TSTF-167, R 00	Rejected	5.0	NI	Not incorporated due to rejection	
TSTF-173, R 00	Approved	3.7	INC		3.7 DOD-44
TSTF-174, R 00	Approved	3.7	INC		3.7 DOD-48
TSTF-196, R 00	Rejected	3.3D 3.6 3.9	NI NI NI	Not incorporated due to rejection	
TSTF-197, R 02	Approved	3.9	NI	Not incorporated due to retention of current license basis	
TSTF-198, R 00	Withdrawn	3.8	NI	Not incorporated due to withdrawal	
TSTF-199, R 00	Withdrawn	3.8	NI	Not incorporated due to withdrawal	
TSTF-200, R 00	Withdrawn	3.8	NI	Not incorporated due to withdrawal	
TSTF-201, R 00	Withdrawn	3.8	NI	Not incorporated due to withdrawal	

TSTF NUMBER	NRC STATUS	ITS SECTION	STATUS(1)	NOTES	DISCUSSION
TSTF-202, R 00	Withdrawn	3.8	NI	Not incorporated due to withdrawal	
TSTF-203, R 00	Withdrawn	3.8	NI	Not incorporated due to withdrawal	
TSTF-204, R 01	Modify	3.8	NI	Not incorporated due to not being approved	
TSTF-205, R 03	Approved	1.0 3.3A 3.3B 3.3C 3.4B	INC NI NI NI NI	Details of the successful test of a channel relay being verified by the change of state of a single contact has not been incorporated since the ANO-1 design of the systems that would be affected by this change do not require this allowance. All required contacts are tested under the existing and proposed test scopes	1.0 DOD-20
TSTF-209, R 01	Approved	3.7	INC		3.7 DOD-07
TSTF-210, R 00	Approved	3.7	INC		3.7 DOD-45
TSTF-211, R 00	Approved	3.3A	IWC	Bases changes proposed in TSTF-211 have been editorially revised for clarity	3.3a DOD-31
TSTF-212, R 01	Approved	3.3A	NI	Not Incorporated due to retention of the ANO-1 current license basis for the affected Surveillance Requirements	
TSTF-213, R 00	Rejected	1.0	NI	Not incorporated due to rejection	
TSTF-214, R 00	Approved	3.9	INC		3.9 DOD-13
TSTF-215, R 00	Withdrawn	3.3B	NI	Not incorporated due to withdrawal	
TSTF-216, R 00	Approved	3.1 3.2	INC INC		3.1 DOD-14 3.2 DOD-16

TSTF NUMBER	NRC STATUS	ITS SECTION	STATUS[1]	NOTES	DISCUSSION
TSTF-217, R 01	Approved	3.3A 3.3B 3.3C	INC IWC NI	TSTF-217 changes were revised in the ITS for LCO 3.3.5, Condition B, to reflect the current license basis. TSTF-217 was not incorporated in ITS 3.3.11 due to retention of the ANO-1 current license basis.	3.3a DOD-02 3.3a DOD-20 3.3b DOD-02
TSTF-218, R 00	Approved	3.3A	INC		3.3a DOD-04
TSTF-219, R 00	Approved	3.4B	NI	Not incorporated due to the revision of the LTOP Required Actions to provide adequate control for inoperabilities associated with the proposed LCO	
TSTF-220, R 00	Approved	3.1	NI	Not incorporated due retention of current license basis	
TSTF-235, R 01	Approved	3.7	INC		3.7 DOD-02
TSTF-248, R 00	Pending	1.0	NI	Not incorporated due to not being approved	
TSTF-249, R 00	Approved	3.1	INC		3.1 DOD-39
TSTF-250, R 00	Modify	4.0 5.0	NI NI	Not incorporated due to not being approved	
TSTF-251, R 00	Rejected	5.0	NI	Not incorporated due to rejection	
TSTF-252, R 00	Pending	5.0	NI	Not incorporated due to not being approved	
TSTF-253, R 00	Approved	3.8	INC		3.8 DOD-03
TSTF-254, R 00	Pending	3.8	NI	Not incorporated due to not being approved	
TSTF-255, R 01	Approved	3.7	INC		3.7 DOD-49
TSTF-256, R 00	Approved	3.1	INC		3.1 DOD-24

TSTF NUMBER	NRC STATUS	ITS SECTION	STATUS(1)	NOTES	DISCUSSION
TSTF-257, R 00	Rejected	3.4B	NI	Not incorporated due to rejection	
TSTF-258, R 04	Approved	5.0	IWC	TSTF-258 has been editorially revised to promote consistency	5.0 DOD-24 5.0 DOD-33 5.0 DOD-42
TSTF-260, R 00	Pending	3.5 3.6 3.7	NI NI NI	Not incorporated due to not being approved	
TSTF-261, R 00	Approved	3.4A	INC		3.4a DOD-16
TSTF-262, R 01	Pending	3.4A	NI	Not incorporated due to not being approved	
TSTF-263, R 03	Approved	3.4A	IWC	TSTF-263 has been revised in its incorporation in ITS 3.4.7 to clarify the Required Actions. This change is considered editorial in nature	3.4a DOD-09 3.4a DOD-12 3.4a DOD-14
TSTF-264, R 00	Approved	3.3A	NI	Not Incorporated due to the incorporation of the ANO-1 current license basis for the affected Surveillance Requirements	
TSTF-265, R 02	Approved	3.4A	INC		3.4a DOD-10
TSTF-266, R 03	Approved	3.3D	NI	Not incorporated due to not being approved	
TSTF-267, R 00	Modify	1.0	NI	Not incorporated due to not being approved	
TSTF-268, R 00	Approved	3.7	INC		3.7 DOD-47
TSTF-269, R 02	Approved	3.6	INC		3.6 DOD-18
TSTF-270, R 01	Withdrawn	1.0	NI	Not incorporated due to withdrawal	

TSTF NUMBER	NRC STATUS	ITS SECTION	STATUS[1]	NOTES	DISCUSSION
TSTF-271, R 01	Approved	3.4B	NI	Not incorporated due to the revision of the LTOP Surveillance Requirements to provide for adequate surveillances with respect to the proposed LCO	
TSTF-272, R 01	Approved	3.9	NI	Not incorporated due to no perceived benefit to ANO-1	
TSTF-273, R 02	Approved	3.0 5.0	INC INC		3.0 DOD-13 5.0 DOD-39
TSTF-274, R 00	Rejected	3.8	NI	Not incorporated due to rejection	
TSTF-276, R 02	Pending	3.8	NI	Not incorporated due to not being approved	
TSTF-277, R 00	Rejected	3.5	NI	Not incorporated due to rejection	
TSTF-278, R 00	Approved	3.8	INC		3.8 DOD-53
TSTF-279, R 00	Approved	5.0	INC		5.0 DOD-11
TSTF-280, R 01	Approved	3.4B	NI	Not incorporated due to incorporation of current license basis	
TSTF-281, R 00	Pending	3.7	NI	Not incorporated due to not being approved	
TSTF-282, R 00	Rejected	3.4A	NI	Not incorporated due to rejection	
TSTF-283, R 02	Pending	3.8	NI	Not incorporated due to not being approved	
TSTF-284, R 02	Pending	1.0 3.1 3.4B 3.7	NI NI NI NI	Not incorporated due to not being approved	
TSTF-285, R 01	Approved	3.4B	NI	Not incorporated due to incorporation of current license basis	

TSTF NUMBER	NRC STATUS	ITS SECTION	STATUS[1]	NOTES	DISCUSSION
TSTF-286, R 01	Modify	3.3A 3.4A 3.4B 3.7 3.8 3.9	NI NI NI NI NI NI	Not incorporated due to not being approved	
TSTF-287, R 04	Pending	3.7	NI	Not incorporated due to not being approved	
TSTF-289, R 00	Approved	3.7	INC		3.7 DOD-08
TSTF-291, R 00	Approved	3.3a	INC		3.3a DOD-17
TSTF-292, R 00	Approved	3.3A	INC		3.3a DOD-21
TSTF-293, R 00	Approved	3.3A	INC		3.3a DOD-23
TSTF-294, R 00	Modify	3.1	NI	Not incorporated due to not being approved	
TSTF-295, R 00	Approved	3.3D	INC		3.3d DOD-18
TSTF-299, R 00	Pending	5.0	INC		5.0 DOD-40
TSTF-300, R 00	Approved	3.8	NI	Not incorporated due to not including NUREG-1430 LCO 3.8.2 in the ANO-1 ITS.	
TSTF-302, R 00	Rejected	4.0	NI	Not incorporated due to rejection	
TSTF-308, R 00	Pending	5.0	INC		5.0 DOD-41
TSTF-312, R 01	Approved	3.9	NI	Not incorporated due to retention of current license basis	

TSTF NUMBER	NRC STATUS	ITS SECTION	STATUS(1)	NOTES	DISCUSSION
TSTF-313, R 00	Pending	3.8	· NI	Not incorporated due to not being approved	
TSTF-316, R 01	Approved	3.5	INC		3.5 DOD-16
TSTF-325, R 00	Approved	3.5	NI	Not incorporated due to the incorporation of the ANO-1 current license basis for the affected Conditions and Required Actions.	
TSTF-327, R 00	Approved	3.3A	NI	Not Incorporated due to the incorporation of the ANO-1 current license basis for the affected Conditions and Required Actions	
TSTF-330, R 00	Pending	3.7	NI	Not incorporated due to not being approved	
TSTF-333, R 01	Pending	3.8	NI	Not incorporated due to not being approved	
TSTF-340, R 02	Pending	3.7	NI	Not incorporated due to not being approved	
TSTF-342, R 00	Pending	3.3A	IWC	TSTF-342 was revised by the incorporation of ANO-1 current license basis	3.3a DOD-30
TSTF-343, R 00	Pending	3.6 5.0	NI NI	Not incorporated due to not being approved	
TSTF-344, R 00	Pending	3.1	INC	Incorporated based on prior NRC approval of similar changes (TSTF-108)	3.1 DOD-02
TSTF-345, R 00	Pending	3.2	INC	Incorporated because this change is necessary to bridge the current license basis to the NUREG-1430 requirements	3.2 DOD-01
TSTF-346, R 00	Pending	3.8	NI	Not incorporated due to not being approved	
TSTF-348, R 00	Approved	5.0	NI	Not incorporated due to retention of current license basis	
TSTF-349, R 00	Pending	3.9	NI	Not incorporated due to not being approved	

TSTF NUMBER	NRC STATUS	ITS SECTION	STATUS(1)	NOTES	DISCUSSION
TSTF-352, R 00	Pending	3.4B	NI	Not incorporated due to not being approved	
TSTF-358, R 01	Pending	3.0	NI	Not incorporated due to not being approved	
TSTF-359, R 01	Pending	3.0	NI	Not incorporated due to not being approved	
TSTF-361, R 00	Pending	3.9	NI	Not incorporated due to not being approved	

(1) NI – Not Incorporated
IWC – Incorporated with Changes
INC - Incorporated

LIST OF BEYOND SCOPE ITEMS

NO.	ITS	CTS	SUMMARY OF CHANGE	CTS DOC	ITS DOD
1	3.2.3 R.A. B.1 Completion Time	3.5.2.6.4	NUREG-1430 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," Required Action B.1 Completion Time has been increased from 2 to 4 hours in the ITS. CTS requires a power reduction if Axial Power Imbalance cannot be restored within 4 hours, however, there is no time limit provided to complete the power reduction.	M8	3.2 DOD-07
2	3.2.4 Completion Times	N/A	A second completion time has been added to NUREG-1430 3.2.4, "QUADRANT POWER TILT (QPT)," Required Action A.1.2.2 requiring a reduction in nuclear overpower based on Reactor Coolant System flow and AXIAL POWER IMBALANCE trip setpoint 10 hours after the last performance of SR 3.2.5.1 when QPT is greater than the steady state limit. Required Action C.1 Completion Time has been revised from 2 hours to 4 hours. Required Action D.1 Completion Time has been revised from 2 hours to 4 hours. An explicit Completion Time is not specified in the CTS.	L1 M1 M6	3.2 DOD-17 3.2 DOD-07 3.2 DOD-07
3	3.4.7	4.27.3	NUREG 3.4.7, "RCS Loops - MODE 5, Loops Filled," LCO b requirement for a specific steam generator level for steam generator OPERABILITY is relocated to the Bases. CTS Surveillance Requirement 4.27.3 specifies a level of ≥20 inches on the startup range. The description of what constitutes an OPERABLE steam generator is described in the associated Bases since there are more options and restrictions than provided in either the NUREG or CTS.	L8	3.4a DOD-18
4	3.4.8	N/A	NUREG-1430 3.4.8 Condition B has been revised by the addition of a Required Action that requires suspension of all activities involving reduction of reactor coolant system volume when two decay heat removal loops are inoperable or one is not in operation. This requirement is not specified in the ANO-1 CTS.	M16	3.4a DOD-25

NO.	ITS	CTS	SUMMARY OF CHANGE	CTS DOC	ITS DOD
5	3.4.11	3.1.2	NUREG-1430 3.4.12, "Emergency Ventilation System," is modified. CTS LTOP requirements provide insufficient guidance on the required actions. The CTS also does not contain surveillances for some of the CTS LCO requirements. The NUREG-1430 3.4.12 Actions and Surveillance Requirements are not totally applicable to the CTS LCO requirements that have been retained.	M3	3.4b DOD-06
6	3.5.2	N/A	NUREG-1430 3.5.2, "ECCS - Operating," has been revised to require a shutdown (in lieu of LCO 3.0.3) for one or more trains inoperable with <100% of the ECCS flow equivalent to a single OPERABLE ECCS train available. CTS does not contain a similar requirement and would require entry into LCO 3.0.3.	A11	3.5 DOD-05
7	3.7.1	3.4.1.2	NUREG-1430 3.7.1, "MSSVs," Figure 3.7.1-1 has been editorially reformatted to be replaced by Table 3.7.1-1 providing limitations for operation with more than one inoperable MSSV per steam generator. CTS does not allow operation with more than 2 inoperable MSSVs.	L1	3.7 DOD-01
8	3.4.13.1	Table 4.1-2 item 6a	NUREG-1430 3.4.13, "RCS Operational Leakage," SR 3.4.13.1 has been modified to require the RCS water inventory balance to be performed "at or near operating pressure." The CTS does not provide this explicit allowance.	L12	3.4b DOD-11
9	6.14	5.5.1	Reference to specific requirements (by Specification number) are deleted from the NUREG 5.5.1.b description of the Offsite Dose Calculation Manual (ODCM). CTS 6.14 also referred to specific requirements. These cross references have been deleted. The ODCM description still refers to the affected reports by name.	A1	5.0 DOD-07

NO.	ITS	CTS	SUMMARY OF CHANGE	CTS DOC	ITS DOD
10	5.2.2.b	Table 6.2-1	NUREG-1430 (incorporating TSTF-258) does not retain Operator staffing. 10 CFR 55.4, related to Operator Licenses, requires a Technical Specification reference to Licensed positions on shift. CTS provides explicit staffing requirements. The ITS contains a reference to 10 CFR 10 CFR 50.54(m)(2)(i).	A5	5.0 DOD-24

LIST OF ITEMS THAT COULD REQUIRE ADDITIONAL NRC RESOURCES

NO.	ITS	CTS	SUMMARY OF CHANGE	CTS DOC	ITS DOD
1	3.2.5	N/A	NUREG-1430 3.2.5, "Power Peaking Factors," Required Action A.1 and A.3 have been revised to require a power reduction to restore linear heat rate to within limits. The associated Completion Time has been revised from 24 hours to 2 hours. Required Action C.1 (ITS R. A. B.1) has been revised to require a power reduction to ≤20% RTP within 4 hours if linear heat rate (LHR) is not restored to within limits within 2 hours. The CTS do not provide guidance for this condition.	M9	3.2 DOD-07 3.2 DOD-23 3.2 DOD-31
2	3.3.3	N/A	NUREG-1430 3.3.3, "Reactor Protection System - Reactor Trip Module," - If two or more reactor trip modules (RTMs) inoperable. If two or more RTMs are inoperable, this change provides specific Required Actions to open the control rod drive trip breakers or remove power from the control rod drive system. LCO 3.0.3 would not require these same actions and could allow the unit to remain in MODE 3 within the applicability. CTS provides no specific actions for inoperable RTMs.	M14	3.3a DOD-20
3	3.3.11	N/A	NUREG-1430 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System," - In addition to the NUREG Action to be <750 PSIG, the ITS provides an option for closing all associated valves in the event Function 3a is inoperable and the Required Actions or associated Completion Times are not met. CTS only requires placing the unit in MODE 3.	M4	3.3c DOD-06

NO.	ITS	CTS	SUMMARY OF CHANGE	CTS DOC	ITS DOD
4	3.3.12	N/A	NUREG-1430 LCO 3.3.12, "Emergency Feedwater Initiation and Control (EFIC) Manual Initiation - Applicability has been revised from MODES 1, 2, and 3 to "When associated EFIC Function is required to be OPERABLE." The CTS does not provide explicit requirements for the Applicability of the EFIC manual trip other than requiring OPERABILITY during startup and power operation. This change ensures the manual initiation backup to the automatic function is available in those MODES in which the Function is required.	M4	3.3c DOD-11
5	3.4.1	N/A	NUREG-1430 LCO 3.4.1, "RCS Pressure Temperature and Flow Departure From Nucleate Boiling Limits," has been revised by relocating the reactor coolant system pressure, temperature and flow departure from nucleate boiling limits to the Core Operating Limits Report, by the addition of a Note to SR 3.4.1.1, and by rewording the Note associated with SR 3.4.1.4. The requirements contained in ITS 3.4.1 were not contained in the CTS.	M10	3.4a DOD-01 3.4a DOD-02 3.4a DOD-03
6	3.4.5	N/A	NUREG-1430 LCO 3.4.5, "RCS Loops - MODE 3," LCO b core outlet temperature is revised from "at least 10°F below" to "sufficiently below saturation temperature to assure subcooling capability." CTS does not contain this limitation.	L7 M15	3.4a DOD-17
7	3.4.10	3.1.1.3.B	NUREG-1430 3.4.10, "Pressurizer Safety Valves," has been modified to reflect the CTS requirement for only one safety in MODE 3 and 4. However, CTS does not provide any actions in this MODE and does not recognize LTOP applicability.	M2	3.4b DOD-03

NO.	ITS	CTS	SUMMARY OF CHANGE	CTS DOC	ITS DOD
8	3.5.1.4	Table 4.1-3 Item 3	NUREG-1430 3.5.1, "Core Flood Tanks," SR 3.5.1.4, " Frequency has been extended from six hours to 12 hours after each solution increase. This sampling Frequency does not exist in the ANO-1 CTS which only states after each makeup. The changes reflect the plant specific time needed to recirculate a core flood tank and to reflect the control room indication available to determine the quantity added to a core flood tank.	L3 M3	3.5 DOD-03
9	3.8.5	N/A	NUREG-1430 3.8.7, "Inverters - Operating," SR 3.8.7.1 Frequency has been revised from 7 days to 31 days. This SR is not required by the CTS.	M1	3.8 DOD-46
10	SR 3.9.3.3	3.8.10	NUREG-1430 3.9.3, "Containment Penetrations," has been revised to include the reactor building purge radiation monitor. NUREG-1430 LCO 3.3.15, "Reactor Building Purge Isolation - High Radiation" was not incorporated in the ITS. The CTS contains a reactor building purge radiation monitor SR with a Frequency of "within 7 days prior to moving fuel." This has been revised to once every 18 months in the ITS.	L3	3.9 DOD-03
11	3.9.5.1	N/A	NUREG-1430 3.9.5, "Decay Heat Removal and Coolant Circulation - Low Water Level," SR 3.9.5.2 has been revised to require verification of correct breaker alignment and indicated power available for each required decay heat removal (DHR) pump. The CTS does not contain this SR.	M1	3.9 DOD-17
12	N/A	5.5.13	NUREG-1430 Specification 5.5.13 description of Diesel Fuel Oil Testing Program has been revised the new fuel oil test of "clear and bright appearance with proper color" to "water and sediment within limits." The CTS does not provide requirements for testing new fuel oil. ANO fuel oil is supplied with added dye which precludes the "clear and bright" test.	M9	5.0 DOD 14
13	3.8	N/A	NUREG-1430 LCOs 3.8.2, 3.8.5, 3.8.8, and 3.8.10 are not incorporated in the ITS. The CTS provides no explicit requirements for these components, treating them as support systems.	N/A	3.8 DOD-17

NO.	ITS	CTS	SUMMARY OF CHANGE	CTS DOC	ITS DOD
14	Various	Various	Parameter values incorporated in the ITS have been revised to reflect the associated analytical values. Allowances for instrument uncertainty have been removed and are proposed to be retained under licensee control.	Various	Various

CTS PAGES NOT INCLUDED IN CTS MARKUP

Pages shown as "DELETED"	66f, 66j, 66m, 66n, 66o, 66p, 66q, 98, 99, 110p, 110q, 110r, 110s, 110t, 110u, 110v, 110w, and 128
Pages not existing in CTS	62 through 65 66k 66l 81 through 91 119 through 125 130 through 139

INTRODUCTION

The purpose of this document is to confirm the results of the Babcock & Wilcox (B&W) Owners Group application of the Technical Specification selection criteria on a plant specific basis for Arkansas Nuclear One -- Unit 1 (ANO-1). Entergy Operations has reviewed this application of selection criteria and confirmed the applicability of the selection criteria to each of the Technical Specifications as reported in BAW-1923, Volume I, "Justification and Background for Technical Specification Improvements" submitted by letter dated February 16, 1987, B&W Owners Group Technical Report 47-1170689-00, "Application of Selection Criteria to the B&W Standard Technical Specifications" submitted by letter dated October 15, 1987, "NRC Staff Review of Nuclear Steam Supply Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications" (Wilgus/Murley letter dated May 9, 1988), and NUREG-1430, "Standard Technical Specifications, Babcock & Wilcox Plants" and applied the criteria to each of the current ANO-1 Technical Specifications. Additionally, in accordance with 10 CFR 50.36 Criterion 4, this confirmation of the application of the selection criteria includes confirmation of the risk insights from probabilistic safety assessment evaluations as applicable to ANO-1.

SELECTION CRITERIA

Entergy Operations has utilized the selection criteria provided in 10 CFR 50.36 (Effective August 18, 1995) to develop the results contained in the attached matrix. Probabilistic Safety Assessment (PSA) insights were utilized and are discussed in the next section of this report. The selection criteria of 10 CFR 50.36 and discussion provided in 60 FR 36959, July 19, 1995, are quoted below.

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

> Discussion of Criterion 1: 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).

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

> Discussion of Criterion 2: 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, which are contained in the SAR, consist of postulated events for which a structure, system or component must meet specified functional goals. They either assume the failure of or present a challenge to the integrity of a fission product barrier.
>
> 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 from the control room. These could also include other features or characteristics that are specifically assumed in the design basis accident and transient analyses even if they

cannot be directly observed in the control room (e.g., moderator temperature coefficient and hot channel factors).

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 low.

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 consequence 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.

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 risk 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 PSA have generally shown to be significant to public health and safety and any other structures, systems, and components that meet this criterion:

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

The Commission recognizes that other structures, systems, and components may meet this criterion. Plant- and design-specific PSAs have yielded valuable insight to unique plant vulnerabilities not fully recognized in safety, design basis accident, or transient analyses. It is the intent of this criterion that those requirements that PSA 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 PSA or risk survey and any available literature on risk insights and PSAs. 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 PSAs in evaluating Technical Specifications related submittals. Further, as a part of the Commission's 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.

PSA INSIGHTS

Introduction and Objectives

The Federal Register (60 FR 36959, July 19, 1995) final rule discussion contains a statement that the NRC expects licensees to utilize any plant-specific PSA or risk survey and any available literature on risk insights and PSAs to strengthen the technical bases for those requirements that remain in Technical Specifications 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.

Those Technical Specifications proposed as being relocated to other plant controlled documents will be maintained under programs subject to the 10 CFR 50.59 review process. These relocated Specifications have been compared to the ANO-1 PSA material with two purposes: 1) to identify if a Specification or topic is addressed by PSA, and 2) if addressed, to judge if the relocated Specification component or topic is risk important. In addition, in some cases risk was judged independent of any specific PSA material. The intent of the review was to provide a supplemental screen to the deterministic criteria.

Assumptions and Approach

Any relocated system or component specifically addressed by PSA material is assumed to participate in core melt or plant risk. The first step in the screening process was to identify those systems and components.

The risk significance of the contribution of an identified system or component was then assessed. PSA data, initiating events, sequence frequencies, fault trees, and event trees were examined to aid in the judgement of the risk significance. In some cases the judgements were clearly supported by the PSA material used. In other cases the judgements were subjective. The assessment was based on plant risk insights and the ANO-1 PSA.

In making the evaluation, judgement was exercised on some components or topics that also require judgement using deterministic criteria. The PSA approach provides a supplemental approach to the use of deterministic criteria but is considered inappropriate for use alone.

RESULTS OF APPLICATION OF SELECTION CRITERIA

The selection criteria from Section 2 were applied to the ANO-1 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 also provided. Entergy Operations will relocate those Specifications identified as not satisfying the criteria to ANO-1 licensee controlled documents whose changes are governed by an appropriate regulatory mechanism, such as 10 CFR 50.59.

CURRENT TS	TITLE	ITS	RETAINED	CRITERION FOR INCLUSION	NOTES
1	**Definitions**	1.1, 3.3.1, 3.6.1, 3.6.2, 3.6.3			This section provides definitions for several defined terms used throughout the remainder of the Technical Specifications. They are provided to improve the meaning of certain terms. As such, direct application of the screening criteria is not appropriate. However, only those definitions for defined terms that remain as a result of application of the screening criteria will remain as definitions in this section of the ITS.
2.1	**Safety Limits, Reactor Core**				
2.1.1	Maximum Fuel Centerline Temperature safety limit	2.1.1.1	Yes	N/A	Application of Technical Specification screening criteria is not appropriate. However, Safety Limits will be included in the ITS as required by 10CFR 50.36.
2.1.2	Departure From Nucleate Boiling Ratio safety limit	2.1.1.2	Yes	N/A	Application of Technical Specification screening criteria is not appropriate. However, Safety Limits will be included in the ITS as required by 10CFR 50.36.
2.1.3	RCS Core Outlet Temperature And Pressure safety limit	2.1.1.3	Yes	N/A	Application of Technical Specification screening criteria is not appropriate. However, Safety Limits will be included in the ITS as required by 10CFR 50.36.
2.2	**Safety Limits - Reactor System Pressure**				
2.2.1	Maximum RCS pressure safety limit	2.1.2	Yes	N/A	Application of Technical Specification screening criteria is not appropriate. However, Safety Limits will be included in the ITS as required by 10CFR 50.36.
2.2.2	Pressurizer Code Safety Valves	3.4.10	Yes	3 & 4	
2.3	**Limiting Safety System Settings, Protective Instrumentation**				
2.3.1	Reactor Protection System Trip Setting Limits	3.3.1	Yes	3 & 4	Application of Technical Specification screening criteria is not appropriate. However, the RPS LSSS have been included as part of the RPS instrumentation Specification, which has been retained since the Functions either actuate to mitigate the consequences of Design Basis Accidents and transients or are retained as directed by the NRC as the Functions are part of RPS.

3.0	**Limiting Conditions For Operation (General)**				
3.0.1	Operational Mode applicability for LCO requirements	3.0.1	Yes	N/A	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operations and Surveillance Requirements. As such, direct application of the Technical Specification screening criteria is not appropriate. However, the general requirements will be retained in the ITS, as modified consistent with NUREG-1430, Rev. 1.
3.0.2	Compliance with the specifications	3.0.2	Yes	N/A	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operations and Surveillance Requirements. As such, direct application of the Technical Specification screening criteria is not appropriate. However, the general requirements will be retained in the ITS, as modified consistent with NUREG-1430, Rev. 1.
3.0.3	Generic Actions for noncompliance	3.0.3	Yes	N/A	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operations and Surveillance Requirements. As such, direct application of the Technical Specification screening criteria is not appropriate. However, the general requirements will be retained in the ITS, as modified consistent with NUREG-1430, Rev. 1.
3.0.4	Entry into Operational Mode restrictions	3.0.4	Yes	N/A	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operations and Surveillance Requirements. As such, direct application of the Technical Specification screening criteria is not appropriate. However, the general requirements will be retained in the ITS, as modified consistent with NUREG-1430, Rev. 1.
3.0.5	Electrical power system inoperabilities		No		Incorporated as Required Actions in ITS 3.8.1 (See 3.8 DOCs L1 and M16)
3.1.1	**Operational Components**				
3.1.1.1	Reactor Coolant Pumps	3.4.4	Yes	2	
3.1.1.2	Steam Generators	3.4.4, 3.4.5	Yes	2 & 3	
3.1.1.3	Pressurizer Safety Valves	3.4.10	Yes	3 & 4	
3.1.1.4	Reactor Internals Vent Valves		No		
3.1.1.5	Reactor Coolant Loops	3.4.4, 3.4.5	Yes	2	
3.1.1.6	Decay Heat Removal	3.4.6, 3.4.7, 3.4.8	Yes	4	
3.1.1.7	Reactor Coolant System Vents		No		Relocated to Technical Requirements Manual (See 3.4a DOCs LA2 and L9)

3.1.2	**Pressurization, Heatup, and Cooldown Limitations**				
3.1.2.1	Hydro Tests	3.4.3	Yes	2	
3.1.2.3	Heatup and Cooldown Rates	3.4.3	Yes	2	
3.1.2.4	Pressurization of Steam Generator Secondary Side		No		Relocated to Technical Requirement Manual. See Appendix A, page 1
3.1.2.5	Pressurizer Heatup and Cooldown Rates		No		Relocated to Technical Requirement Manual. See Appendix A, page 3
3.1.2.9	Core Flood Tank Discharge Valve Controls	3.4.11	Yes	2	
3.1.2.10	HPI MOV LTOP Control	3.4.11	Yes	2	
3.1.2.11	Reactor Coolant System Not Solid	3.4.11	Yes	2	
3.1.3	**Minimum Conditions For Criticality**				
3.1.3.1	Reactor Coolant System Temperature	3.4.2	Yes	2	
3.1.3.2	Reactor Coolant Temperature	3.4.3	Yes	2	
3.1.3.4	Pressurizer Level	3.4.9	Yes	2 & 4	
3.1.3.5	Regulating Rod Position Limits	3.1.5, 3.1.8, 3.1.9	Yes	2, 3, & 4	
3.1.3.6	Emergency Powered Pressurizer Heaters	3.4.9	Yes	2 & 4	
3.1.4	**Reactor Coolant System Activity**				
3.1.4.1.a	Total Specific Activity	3.4.12	Yes	2	
3.1.4.1.b	The I-131 dose equivalent of the radioiodine activity	3.4.12	Yes	2	
3.1.5	**Chemistry**				
3.1.5.1	Reactor Coolant Contaminant limits		No		Relocated to Technical Requirement Manual (See 3.4a DOCs LA2 and L10)
3.1.6	**Leakage**				
3.1.6.1	Total RCS Leakage	3.4.13	Yes	2 & 4	
3.1.6.2	Unidentified RCS Leakage	3.4.13	Yes	2 & 4	
3.1.6.3.a	RCS Strength Boundary Leakage	3.4.13	Yes	2 & 4	
3.1.6.3.b	Steam Generator Tube Leakage	3.4.13	Yes	2 & 4	
3.1.6.7	Reactor Coolant Leak Detection Instrumentation	3.4.15	Yes	1 & 4	
3.1.6.8	Returnable RCS Leakage		No		Relocated to Technical Requirement Manual (See 3.4b DOCs LA2 and M5)
3.1.6.9	Pressure Isolation Valve Leakage	3.4.14	Yes	4	

3.1.7	**Moderator Temperature Coefficient of Reactivity Specification**				
3.1.7.1	Moderator Temperature Coeficient	3.1.3	Yes	2 & 4	
3.1.8	**Low Power Physics Testing Restrictions**				
3.1.8.1	Reactor Protective System Requirements	3.1.9	Yes	N/A	
3.1.8.2	Startup Rate Rod Withdrawal Hold	3.1.9	Yes	N/A	
3.1.8.3	Minimum Reactor Coolant Temperature and Shutdown Margin	3.1.8, 3.1.9	Yes	4	
3.1.9	**Control Rod Operation**				
3.1.9.1	Concentration of Dissolved Gases in Reactor Coolant System		No		Relocated to the Technical Requirement Manual (See 3.1 DOC LA1)
3.1.9.2	Allowable Combinations of Pressure and Temperature for Control Rod Operations		No		Relocated to Technical Requirement Manual (See 3.1 DOCs LA1 and L14)
3.2	**Makeup and Chemical Addition Systems**				
3.2.1.1	Makeup Pump Operability		No		Relocated to Technical Requirement Manual (See 3.5 DOCs LA3 and L11)
3.2.1.2	Source of Concentrated Boric Acid Solution		No		Relocated to Technical Requirement Manual (See 3.5 DOCs LA3 and L11)
3.3	**Emergency Core Cooling, Reactor Building Emergency Cooling and Reactor Building Spray Systems**				
3.3.1.A	Reactor Building Spray Pump	3.6.5	Yes	3 & 4	
3.3.1.B	Reactor Building Emergency Cooling	3.6.5	Yes	3 & 4	
3.3.1.C	Service Water Pumps	3.7.7	Yes	3 & 4	
3.3.1.D	Low Pressure Injection (LPI) Pumps	3.5.2, 3.5.3	Yes	3 & 4	
3.3.1.E	Low Pressure Injection Coolers	3.5.2, 3.5.3	Yes	3 & 4	
3.3.1.F	Borated Water Storage Tank (BWST) Level Instrument Channels	3.3.15	Yes	3 & 4	Table 3.3.5-1, PAM # 15
3.3.1.G	BWST Level and Concentration	3.5.4	Yes	3 & 4	
3.3.1.H	Reactor Building Emergency Sump Isolation Valves		No		Relocated to Bases for ITS 3.5.2 and 3.5.3. See 3.5 DOC LA1.

3.3.1.I	Engineered Safety Features Valves	3.5.2, 3.5.3, 3.5.4, 3.6.5, 3.7.7	Yes	3 & 4	
3.3.2.A	High Pressure Injection (Makeup) Pumps	3.5.2	Yes	3 & 4	
3.3.2.B	Engineered Safety Features Valves (HPI)	3.5.2	Yes	3 & 4	
3.3.3.A	Core Flooding Tank Level and Pressure	3.5.1	Yes	3 & 4	
3.3.3.B	Core Flooding Tank Boron Concentration	3.5.1	Yes	3 & 4	
3.3.3.C	Core Flooding Tank Discharge	3.5.1	Yes	3 & 4	
3.3.3.D	Core Flooding Tank Pressure and Instrument Channels		No		Relocated to Technical Requirement Manual (See 3.5 DOC LA3)
3.3.4.A	Reactor Building Spray Pump and Emergency Cooler with Reactor Critical	3.6.5, 3.6.6	Yes	3 & 4	
3.3.4.C	Sodium Hydroxide Tank Manual Valve Status	3.6.6	Yes	3	
3.3.4.D	Engineered Safety Features Valves and Interlocks	3.5.1, 3.5.2, 3.5.3, 3.5.4, 3.6.5, 3.6.6, 3.7.7	Yes	3 & 4	
3.4	**Steam and Power Conversion**				
3.4.1.1	Two Steam Generators	3.4.4, 3.4.5	Yes	2 & 3	
3.4.1.2	Main Steam Safety Valves	3.7.1	Yes	3 & 4	
3.4.1.3	Condensate Storage Tank Level	3.7.6	Yes	4	
3.4.1.5	Main Steam Line Block Valve & Main Feedwater Isolation Valve	3.7.2, 3.7.3	Yes	3 & 4	
3.4.3	Two EFW trains shall be operable as follows:	3.7.5	Yes	3 & 4	

3.5.1	**Operational Safety Instrumentation**				
3.5.1.4	RPS key operated shutdown bypass switch		No		Relocated to Technical Requirement Manual. See Appendix A, page 5.
3.5.1.12	Containment High Range Radiation Monitor	3.3.15	Yes	3 & 4	Table 3.3.15-1, PAM # 9
3.5.1.13	Control Room ventilation radiation monitoring channel	3.3.16	Yes	3	
3.5.1.14	The Main Steam Line Radiation Monitoring Instrumentation.		No		Relocated to ODCM and SAR (See 3.3d DOC LA2)
3.5.1.15	EFIC System Initiate Functions	3.3.11	Yes	3 & 4	
3.5.1.16	EFIC Steam Generator Isolation System	3.3.11	Yes	3 & 4	
3.5.1-1	Reactor Protection System (RPS) Functions	3.3.1, 3.3.2, 3.3.3, 3.3.4, 3.3.9, 3.3.10	Yes	3 & 4	
3.5.1-1	Engineered Safeguards Actuation System (ESAS)	3.3.5, 3.3.6, 3.3.7, 3.3.15	Yes	3 & 4	
3.5.1-1	Emergency Feedwater Initiation and Control (EFIC)	3.3.11, 3.3.12, 3.3.13, 3.3.14, 3.3.15	Yes	3 & 4	
3.5.1-1	Decay heat removal system isolation valve automatic closure and interlock system	3.4.14	Yes	4	
3.5.1-1	Pressurizer level channels	3.3.15	Yes	3 & 4	Table 3.3.15-1 PAM # 13
3.5.1- 1	Emergency feedwater flow channels	3.3.15	Yes	3 & 4	Table 3.3.15-1 PAM # 17
3.5.1-1	RCS subcooling margin monitors		No		See 3.3d DOC L15
3.5.1-1	Electromatic relief valve flow monitor		No		See 3.3d DOC L15
3.5.1-1	Electromatic relief block valve position indicator		No		See 3.3d DOC L15
3.5.1-1	Pressurizer code safety valve flow monitor		No		See 3.3d DOC L15
3.5.1-1	Degraded voltage monitoring	3.3.8	Yes	3 & 4	
3.5.1- 1	Containment high range radiation monitoring requirements	3.3.15	Yes	3 & 4	Table 3.3.15-1 PAM # 9

3.5.1-1	Containment pressure - high range LCO	3.3.15	Yes	3 & 4	Table 3.3.15-1 PAM # 7
3.5.1-1	Containment water level - wide range LCO	3.3.15	Yes	3 & 4	Table 3.3.15-1 PAM # 6
3.5.1-1	In core thermocouples (core-exit thermocouples)	3.3.15	Yes	3 & 4	Table 3.3.15-1 PAM # 16
3.5.1-1	Control Room Radiation Monitors	3.3.16	Yes	3	
3.5.1-1	Reactor Vessel Level Monitoring System	3.3.15	Yes	3 & 4	Table 3.3.15-1 PAM # 5
3.5.1-1	Hot Leg Level Measurement System (HLLMS)	3.3.15	Yes	3 & 4	Table 3.3.15-1 PAM # 3
3.5.1-1	Main Steam Line Radiation Monitors		No		Relocated to ODCM and SAR (See 3.3d DOC LA2)
3.5.2	**Control Rod Group and Power Distribution Limits**				
3.5.2.1	Available Shutdown Margin During Power Operation	3.1.5, 3.2.1	Yes	2, 3, & 4	
3.5.2.4	Quadrant Power Tilt Restrictions	3.1.8, 3.2.4	Yes	2	
3.5.2.5	Control Rod Position	3.1.8, 3.1.9, 3.2.1, 3.2.2	Yes	2 & 4	
3.5.2.6	Reactor Power Imbalance	3.1.8, 3.2.3	Yes	2	
3.5.2.7	Control Rod Drive Patch Panel Restrictions		No		Relocated to SAR (See 3.2 DOC LA1)
3.5.3	**Safety Features Actuation System Setpoints**	3.3.5	Yes	3 & 4	
3.5.4	**Incore Instrumentation**		No		Relocated to Bases for 3.2.3 and 3.2.4 (See 3.2 DOCs LA1 and M18)
3.6	**Reactor Building**				
3.6.1	Reactor Building Operability	3.6.1	Yes	3 & 4	
3.6.2	Reactor Building Integrity Restrictions When The RCS Is Open To Atmosphere		No		Deleted (See 3.6 DOC L4)
3.6.4	Reactor Building Internal Pressure Limit	3.6.4	Yes	2 & 4	

3.7	**Auxiliary Electrical Systems**				
3.7.1.A	Offsite Power Sources	3.8.1	Yes	3 & 4	
3.7.1.B	Distribution System Operability	3.8.6	Yes	3 & 4	
3.7.1.C	Diesel Generator Operability	3.8.1, 3.8.2	Yes	3 & 4	
3.7.1.F	Off-site power undervoltage and protective relaying interlocks	3.8.1	Yes	3 & 4	
3.7.1.G	Selective load-shed features associated with Startup Transformer No.2	3.8.1	Yes	3 & 4	
3.7.3	125 VDC Electrical Power Subsystems	3.8.3	Yes	3 & 4	
3.7.4	Battery Cell Parameters	3.8.4	Yes	3	
3.8	**Fuel Loading And Refueling**				
3.8.1	Radiation Level Monitors for Refueling		No		Relocated to Technical Requirement Manual. See Appendix A, page 7.
3.8.2	Neutron Flux Monitoring Instrumentation	3.9.2	Yes	4	
3.8.3	Decay heat removal operation during fuel handling	3.9.4, 3.9.5	Yes	4	
3.8.4	Boron Concentration	3.9.1	Yes	2	
3.8.5	Direct Communications between control room and refueling personnel		No		Relocated to Technical Requirement Manual. See Appendix A, page 9.
3.8.6	Reactor Building Equipment hatch and Personnel Hatch	3.9.3, 3.9.6	Yes	2 & 4	
3.8.7	Penetration Isolation Valves	3.9.3	Yes	4	
3.8.8	Fuel Assembly Separation		No		Relocated to Technical Requirement Manual. See Appendix A, page 11.
3.8.10	Reactor Building Purge Isolation Sys	3.9.3	Yes	4	
3.8.11	Minimum Time After Shutdown		No		Relocated to Technical Requirement Manual (See 3.9 DOC LA2)
3.8.12	Severe Weather		No		Relocated to Technical Requirement Manual. See Appendix A, page 13.
3.8.13	Spent Fuel Cask Movement		No		Relocated to Technical Requirement Manual. See Appendix A, page 15.
3.8.14	Heavy Loads		No		Relocated to Technical Requirement Manual. See Appendix A, page 17
3.8.15	Fuel Enrichment in Spent Fuel Pool	4.3.1.1.A	Yes	N/A	
3.8.16	Burnup Restriction on Storage in Region 2 of Spent Fuel Pool	3.7.14	Yes	2	
3.8.17	Boron Concentration in Spent Fuel Pool	3.7.13	Yes	2	
3.8.18	Control Room Emergency Air Conditioning and Emergency Filtration System	3.7.9, 3.7.10	Yes	3 & 4	

3.9	**Control Room Emergency Air Conditioning And Isolation System**				
3.9.1	Control Room Emergency Air Conditioning System	3.7.10	Yes	3 & 4	
3.9.2	Control Room Emergency Ventilation System	3.7.9	Yes	3 & 4	
3.10	**Secondary System Activity**	3.7.4	Yes	2 & 4	
3.11	**Emergency Cooling Pond**				
3.11.1	Emergency Cooling Pond Operability	3.7.8	Yes	3	
3.12	**Miscellaneous Radioactive Materials Sources**		No		Relocated to Technical Requirement Manual (See 3.7 DOC LA3)
3.13	**Penetration Room Ventilation System**				
3.13.1	Penetration Room Ventilation System Operability	3.7.11	Yes	3 & 4	
3.14	**Hydrogen Recombiners**				
3.14.1	Hydrogen Recombiner Operability	3.6.7	Yes	3	
3.14.3	Hydrogen concentration instruments	3.3.15	Yes	3 & 4	Table 3.3.15-1 PAM # 10
3.15	**Fuel Handling Area Ventilation**				
3.15.1	Fuel Handling Area Ventilation Operability	3.7.12	Yes	3	
3.16	**Shock Suppressors (Snubbers)**		No		Relocated to Technical Requirement Manual. See Appendix A, page 19
3.22	**Reactor Building Purge Filtration System**		No		Relocated to Technical Requirement Manual. See Appendix A, page 21
3.23	**Reactor Building Purge Valves**				
3.23.1	Reactor Building Purge Valve Requirements	3.6.3	Yes	3 & 4	
3.24	**Explosive Gas Mixture**				
3.24.1	Waste Gas Decay Tank Explosive Gas Limits	5.5.12	Yes	N/A	In accordance with agreements reached between the Industry and NRC, this specification will be retained as a program in Administrative Controls. See 5.0 DOC LA5.
3.25	**Radioactive Effluents**				
3.25.1	Radioactive Liquid Effluent Release Limits	5.5.12	No		In accordance with agreements reached between the Industry and NRC, this specification will be retained as a program in Administrative Controls. See 5.0 DOC LA5.
3.25.2	Waste Gas Storage Tank Curie Limit	5.5.12	No		In accordance with agreements reached between the Industry and NRC, this specification will be retained as a program in Administrative Controls. See 5.0 DOC LA5.

4.0	**Surveillance Requirements**				
4.0.1	Operational Modes	SR 3.0.1	Yes	N/A	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operations and Surveillance Requirements. As such, direct application of the Technical Specification screening criteria is not appropriate. However, the general requirements will be retained in the ITS, as modified consistent
4.0.2	Surveillance Interval and Extension	SR 3.0.2	Yes	N/A	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operations and Surveillance Requirements. As such, direct application of the Technical Specification screening criteria is not appropriate. However, the general requirements will be retained in the ITS, as modified consistent
4.0.4	Entry into Operational Mode restrictions	SR 3.0.4	Yes	N/A	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operations and Surveillance Requirements. As such, direct application of the Technical Specification screening criteria is not appropriate. However, the general requirements will be retained in the ITS, as modified consistent
4.0.5	ASME Code Class 1, 2, 3 Components IST	5.5.8	Yes	N/A	This Specification provides generic guidance applicable to one or more Specifications. The information is provided to facilitate understanding of Limiting Conditions for Operations and Surveillance Requirements. As such, direct application of the Technical Specification screening criteria is not appropriate. However, the general requirements will be retained in the ITS, as modified consistent
4.0.5	ASME Code Class 1, 2, 3 Components ISI		No		See 5.0 DOD A5
4.1	**Operational Safety Items**				
4.1.d	Power Distribution Map	3.2.5	Yes	2	
4.1- 1.38	Sodium Hydroxide Tank Level Indicator		No		Relocated to the Technical Requirement Manual. See Appendix A, page 23. Split Criteria applied to implied LCO. There is no explicit LCO requirement associated with this surveillance requirement.
4.1- 2.5	Refueling System Interlocks - Functioning Test		No		Relocated to the Technical Requirement Manual. See Appendix A, page 25 of Split Report. Split Criteria applied to implied LCO. There is no explicit LCO requirement associated with this surveillance requirement.
4.1- 2.10	Spent Fuel Cooling System - Functioning Test		No		Relocated to the Technical Requirement Manual. See Appendix A, page 25. Split Criteria applied to implied LCO. There is no explicit LCO requirement associated with this surveillance requirement.

4.1- 3.1.f	Reactor Coolant Boron Concentration		No		Relocated to the Technical Requirement Manual. See Appendix A, page 27. Split Criteria applied to implied LCO. There is no explicit LCO requirement for non-MODE 6 operating conditions associated with this surveillance requirement.
4.1- 3.5.a	Secondary Coolant - Gross Radioiodine		No		Relocated to the Technical Requirement Manual. See Appendix A, page 27. Split Criteria applied to implied LCO. There is no explicit LCO requirement associated with this surveillance requirement.
4.6	**Auxiliary Electrical System Tests**				
4.6.3	Emergency Lighting		No		Relocated to the Technical Requirement Manual. See Appendix A, page 29. Split Criteria applied to implied LCO. There is no explicit LCO requirement associated with this surveillance requirement.
4.7	**Reactor Control Rod System Tests**				
4.7.2	Control Rod Program Verification		No		Relocated to the SAR. See Appendix A, page 31. Split Criteria applied to implied LCO. There is no explicit LCO requirement associated with this surveillance requirement.
4.15	**Augmented Inservice Inspection Program For High Energy Lines Outside of Containment**		No		Relocated to the Inservice Inspection Program. See Appendix A, page 33. Split Criteria applied to implied LCO. There is no explicit LCO requirement associated with this surveillance requirement.
5.0	**Design Features**				
5.1	Site	4.1	Yes	N/A	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Design Features will be included in the ITS as required by 10 CFR 50.36. Individual Items are addressed in DOCs for 4.0.
5.2	Reactor Building		No		Application of Technical Specification screening criteria is not appropriate. However, specific portions of Design Features will be included in the ITS as required by 10 CFR 50.36. Individual Items are addressed in DOCs for 4.0.
5.3	Reactor	4.2	Yes	N/A	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Design Features will be included in the ITS as required by 10 CFR 50.36. Individual Items are addressed in DOCs for 4.0.
5.4	New and Spent Fuel Storage Facilities	4.3	Yes	N/A	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Design Features will be included in the ITS as required by 10 CFR 50.36. Individual Items are addressed in DOCs for 4.0.

6.0	**Administrative Controls**				
6.1	Responsibility	5.1	Yes	N/A	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Administrative Controls will be included in the ITS as required by 10 CFR 50.36. Individual Items are addressed in DOCs for 5.0.
6.2	Organization	5.2	Yes	N/A	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Administrative Controls will be included in the ITS as required by 10 CFR 50.36. Individual Items are addressed in DOCs for 5.0.
6.3	Facility Staff Qualifications	5.3	Yes	N/A	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Administrative Controls will be included in the ITS as required by 10 CFR 50.36. Individual Items are addressed in DOCs for 5.0.
6.7	Safety Limit Violation	2.0	Yes	N/A	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Administrative Controls will be included in the ITS as required by 10 CFR 50.36. Individual Items are addressed in DOCs for 2.0.
6.8	Procedures and Programs	5.4, 5.5	Yes	N/A	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Administrative Controls will be included in the ITS as required by 10 CFR 50.36. Individual Items are addressed in DOCs for 5.0.
6.10	Radiation Protection Program		No		See 5.0 DOC A5
6.11	High Radiation Area	5.7	Yes	N/A	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Administrative Controls will be included in the ITS as required by 10 CFR 50.36. Individual Items are addressed in DOCs for 5.0.
6.12	Reporting Requirements				
6.14	Offsite Dose Calculation Manual (ODCM)	5.6	Yes	N/A	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Administrative Controls will be included in the ITS as required by 10 CFR 50.36. Individual Items are addressed in DOCs for 5.0.

OL	Operating License				
OL 2.C.5	Operating License Condition -- System Integrity	5.5.2	Yes	2	The Technical Specification selection criteria have been applied to this operating License Condition since this requirement was included as a Specification in the old STS (NUREG-0123). The results of this evaluation have concluded that this condition meets the inclusion criteria and has been retained as a program in the ITS.
OL 2.C.6	Operating License Condition -- Iodine Monitoring	5.5.3	Yes	4	The Technical Specification selection criteria have been applied to this operating License Condition since this requirement was included as a Specification in the old STS (NUREG-0123). The results of this evaluation have concluded that this condition meets the inclusion criteria and has been retained as a program in the ITS.
OL 2.C.7	Operating License Condition -- Secondary Water Chemistry Monitoring	5.5.10	Yes	4	The Technical Specification selection criteria have been applied to this operating License Condition since this requirement was included as a Specification in the old STS (NUREG-0123). The results of this evaluation have concluded that this condition meets the inclusion criteria and has been retained as a program in the ITS.

APPENDIX A

3.1.2.4 STEAM GENERATOR P/T LIMITS

LCO Statement:

3.1.2.4 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100F.

LCO Related Requirements:

3.1.2.6

Discussion:

The limitations on steam generator pressure and temperature provide protection against nonductile failure of the secondary side (shell) of the steam generator. These limits are calculated using the ASME code for Class A components and are considered to be conservative. An engineering evaluation of the continued structural integrity of the steam generators is required if these limits are exceeded. The actions associated with the limits and evaluation are located in TS 3.1.2.6.

Comparison to Screening Criteria:

Criterion 1

The steam generator P/T limits do not constitute an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

Steam generator P/T limits are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier. This Technical Specification specifies limits on process variables consistent with the structural analysis results. These limits, however, do not reflect initial condition assumptions in the DBA.

Criterion 3

Steam generator P/T limits are not 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.

Criterion 4

PSA does not address steam generator P/T limits or brittle fracture of the shell. Except for wet lay-up, the conditions to permit violation of the limits are virtually nonexistent. It is inferred that the risk due to shell failure is small.

Conclusion:

Since the screening criteria have not been satisfied, the steam generator P/T limit requirements may be relocated to licensee controlled documents outside the Technical Specifications.

3.1.2.5 PRESSURIZER HEATUP AND COOLDOWN LIMITS

LCO Statement:

3.1.2.5 The pressurizer heatup and cooldown rates shall not exceed 100F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430F.

LCO Related Requirements:

3.1.2.6

Discussion:

The heatup and cooldown rates and differential temperature limitation are placed on the pressurizer to prevent non-ductile failure and assure compatibility of operation with the fatigue analysis performed. The limits meet the requirements given in ASME Section III, Appendix G. These limitations are consistent with structural analysis results and are considered to be conservative. An engineering evaluation of the continued structural integrity of the pressurizer is required if these limits are exceeded. The actions associated with the limits and evaluation are located in TS 3.1.2.6.

Comparison to Screening Criteria:

Criterion 1

Pressurizer heatup and cooldown rates and the spray fluid temperature limitation are not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

Pressurizer heatup and cooldown rates and the spray fluid temperature limitation are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

Pressurizer heatup and cooldown rates and temperature limitation are not 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.

Criterion 4

The ANO-1 PSA does not address pressurizer P/T limits, temperature limits, or the integrity of the pressurizer (with the exception of the ERV and SRVs). This heatup and cooldown rate is considered to be a non-risk contributor to the core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the pressurizer P/T and spray fluid temperature limit requirements may be relocated to licensee controlled documents outside the Technical Specifications.

3.5.1.4 RESTRICTION ON USE OF SHUTDOWN BYPASS KEY SWITCH DURING POWER OPERATION

LCO Statement:

3.5.1.4 The key operated shutdown bypass switch associated with each reactor protection channel shall not be used during reactor power operation except during channel testing.

LCO Related Requirements:

NA

Discussion:

The shutdown bypass switch enables the power/imbalance/flow, power/pump, low pressure, and pressure/temperature trips to be bypassed allowing Control Rod Drive tests to be performed after the reactor has been shut down and depressurized below the low reactor coolant pressure trip point. This bypass is indicated on the plant annunciator, the cabinet alarm lamp panel, and internally in the cabinet. Before the bypass may be initiated, a high pressure trip bistable, which is incorporated in the shutdown bypass circuitry, must be manually reset. The setpoint of the high pressure bistable (associated with shutdown bypass) is set below the low pressure trip point. If pressure is increased with the bypass initiated, the channel will trip when the high pressure bistable associated with the shutdown bypass trips. Additionally, trip protection is provided while in shutdown bypass for reactivity addition accidents at low system temperature and pressure. During any such accident, any safety or regulating rods that are withdrawn will be automatically inserted in the core if the flux level exceeds the bistable setpoint. The use of the shutdown bypass key switch is under administrative control.

Comparison to Screening Criteria:

Criterion 1

The RPS key operated shutdown bypass switch is not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

The RPS key operated shutdown bypass switch is not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

The RPS key operated shutdown bypass switch is not 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.

Criterion 4

The use of the shutdown bypass switch while at power would require an act of commission by the operator. Acts of commission are not considered credible in the PSA model. Therefore, the ANO-1 PSA does not consider the status of the key operated shutdown bypass switch. Actuating the keyswitch during reactor power operation with no RPS channel testing in progress would result in a channel trip. Use of the keyswitch is considered a non-significant contributor to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the key operated shutdown bypass switch requirements may be relocated to licensee controlled documents outside the Technical Specifications.

3.8.1 RADIATION MONITORING INSTRUMENTATION DURING FUEL LOADING AND REFUELING

LCO Statement:

3.8.1 Radiation levels in the reactor building refueling area shall be monitored by instrument RE-8017. Radiation levels in the spent fuel storage area shall be monitored by instrument RE-8009. If any of these instruments become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.

LCO Related Requirements:

3.8.9

Discussion:

Radiation monitors RE-8017 and RE-8009 are permanently installed in areas of personnel activity during fuel loading, refueling and fuel handling and provide an alarm locally and in the Control Room when triggered. When either is inoperative, the local radiation coverage and alarm functions are provided by portable survey instrumentation.

Comparison to Screening Criteria:

Criterion 1

Radiation monitoring in the reactor building refueling area and spent fuel storage area is not used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

Radiation monitoring in the reactor building refueling area and spent fuel storage area is not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

Radiation monitoring in the reactor building refueling area and spent fuel storage area is not 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.

Criterion 4

Radiation monitoring in the reactor building refueling area and spent fuel storage area is not addressed in the ANO-1 PSA and is not credited in any accident analysis and is considered to be non-risk significant with respect to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the radiation monitoring instrumentation during fuel loading and refueling requirements may be relocated to a licensee controlled document outside of the Technical Specifications.

3.8.5 DIRECT COMMUNICATIONS DURING CHANGES IN CORE GEOMETRY

LCO Statement:

3.8.5 Direct communications between the control room and the refueling personnel in the reactor building shall exist whenever changes in core geometry are taking place.

LCO Related Requirements:

3.8.9

Discussion:

Communications between the control room personnel and personnel performing core alterations is maintained to ensure that personnel can be promptly informed of significant changes in the plant status or core reactivity condition during refueling. The communications allow for coordination of activities that require interaction between the control room and refueling personnel.

Comparison to Screening Criteria:

Criterion 1

Direct communications during changes in core geometry is not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

Direct communications during changes in core geometry is not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

Direct communications during changes in core geometry is not 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.

Criterion 4

Communications during changes in core geometry while in shutdown mode is beyond the scope of the at power PSA model and is considered to be non-risk significant with respect to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, direct communications during changes in core geometry requirements may be relocated to other licensee controlled documents outside the Technical Specifications.

3.8.8 MINIMUM SEPARATION BETWEEN FUEL HANDLING BRIDGES

LCO Statement:

3.8.8 When two irradiated fuel assemblies are being moved simultaneously by the bridges within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times.

LCO Related Requirements:

3.8.9

Discussion:

When being moved, irradiated fuel assemblies should not be brought close to each other due to the possibility of a criticality accident or, more likely, cladding damage by contact. In normal use, it is physically impossible for fuel assemblies being moved with the fuel transfer canal bridges to be within 10 feet of each other. This restriction considers abnormal use of this equipment.

Comparison to Screening Criteria:

Criterion 1

The separation requirement when moving irradiated fuel assemblies is not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

The separation requirement when moving irradiated fuel assemblies is not addressed in the ANO-1 PSA and is not credited in any accident analysis. The separation requirement is considered to be non-risk significant with respect to core damage and offsite releases because the physical dimensions of the fuel bridges make it impossible for fuel assemblies to be within 10 feet of each other while being handled.

Criterion 3

The separation requirement when moving irradiated fuel assemblies is not 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.

Criterion 4

The separation requirement when moving irradiated fuel assemblies is not addressed in the ANO-1 PSA and is not credited in any accident analysis. The separation requirement is considered to be non-risk significant with respect to core damage and offsite releases because the physical dimensions of the fuel bridges make it impossible for fuel assemblies to be within 10 feet of each other while being handled.

Conclusion:

Since the screening criteria have not been satisfied, the separation requirements when moving irradiated fuel assemblies may be relocated to other licensee controlled documents outside the Technical Specifications

3.8.12 FUEL HANDLING OPERATIONS UNDER TORNADO WATCH

LCO Statement:

3.8.12 All fuel handling in the Auxiliary Building shall cease upon notification of the issuance of a tornado watch for Pope, Yell, Johnson, or Logan counties in Arkansas. Fuel handling operations in progress will be completed to the extent necessary to place the fuel handling bridge and crane in their normal parked and locked position. The provisions of Specification 3.0.3 are not applicable.

LCO Related Requirements:

NA

Discussion:

When a tornado watch has been declared by the National Weather Service for the vicinity of ANO then fuel handling operations in the Auxillary Building must be ceased and the related equipment placed in a safe configuration. These actions are part of the requirements for responding to High Winds/Tornado/Thunderstorm portion of the Natural Emergencies procedure.

Comparison to Screening Criteria:

Criterion 1

The restriction on fuel handling during a local tornado watch is not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

The restriction on fuel handling during a local tornado watch is not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

The restriction on fuel handling during a local tornado watch is not 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.

Criterion 4

The restriction on fuel handling during a local tornado watch is not addressed in the ANO-1 PSA and is not credited in any accident analysis and is considered to be non-risk significant with respect to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the requirements for moving irradiated fuel during a declared tornado watch in the local area, may be relocated to other licensee controlled documents outside the Technical Specifications.

3.8.13 REQUIREMENTS FOR MOVEMENT OF SPENT FUEL SHIPPING CASKS

LCO Statement:

3.8.13 No loaded spent fuel shipping cask shall be carried above or into the Auxiliary Building equipment shaft unless atmospheric dispersion conditions are equal to or better than those produced by Pasquill Type D stability accompanied by a wind velocity of 2 m/sec. In addition, the railroad spur door of the Turbine Building shall be closed and the fuel handling area ventilation system shall be in operation. The provisions of Specification 3.0.3 are not applicable.

LCO Related Requirements:

NA

Discussion:

When ANO was originally designed, the movement of a loaded spent fuel shipping cask into or above the Auxiliary Building equipment shaft, (the "train bay" west end), was to be limited to periods of certain atmospheric dispersion conditions and with the rail spur door shut and the fuel handling area ventilation operating. ANO has recently installed the capability for dry storage of 5-year-cooled spent fuel assemblies which has altered the calculation of offsite dose from a shipping cask drop.

A new assessment of the loaded spent fuel shipping cask drop event has been prepared concurrent with the pursuit of dry storage for spent fuel at ANO. The original ANO-1 SAR dropped-cask analysis assumed 100 day cooled fuel and an offsite dose calculation based on non-SRP methodology. With 5 year cooled fuel, using either SRP or original methodology the offsite dose is within 10CFR100 requirements even on a calm day, (i.e. Pasquill type D dispersion conditions with wind velocity of 2 m/sec. are not required).

Comparison to Screening Criteria:

Criterion 1

These restrictions on movement of a loaded spent fuel shipping cask are not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

These restrictions on movement of a loaded spent fuel shipping cask are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

These restrictions on movement of a loaded spent fuel shipping cask are not 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.

Criterion 4

The restrictions on movement of a loaded spent fuel shipping cask are not addressed in the ANO-1 PSA and are considered to be non-risk significant with respect to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the requirements for movement of a loaded spent fuel shipping cask may be relocated to other licensee controlled documents outside the Technical Specifications

3.8.14 2000 POUND LOAD LIMIT OVER SPENT FUEL POOL

LCO Statement:

3.8.14 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool. The provisions of Specification 3.0.3 are not applicable.

LCO Related Requirements:

NA

Discussion:

This Specification ensures that no loads heavier than the weight of a single spent fuel assembly plus the tool for moving the assembly will be carried over fuel stored in the spent fuel pool. In the event that the load is dropped, the activity released is limited to that assumed in the fuel handling accident analysis. This also prevents any possible distortion of fuel assemblies in the storage racks from resulting in a critical configuration.

Comparison to Screening Criteria:

Criterion 1

This heavy load restriction is not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

This heavy load restriction is not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

This heavy load restriction is not 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.

Criterion 4

The restrictions on movement of excess loads over fuel assemblies in the storage pool are not addressed in the ANO-1 PSA but are considered to be non-risk significant with respect to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the 2000 pound load over the spent fuel pool requirements may be relocated to other licensee controlled documents outside the Technical Specifications.

3.16 SHOCK SUPPRESSORS (SNUBBERS)

LCO Statement:

3.16.1 With one or more applicable snubbers inoperable, within 72 hours either:

LCO Related Requirements:

3.16 Applic, 3.16 Obj, 3.16.1 a, 3.16.1 b, 3.16.1 c, Bases & 4.16

Discussion:

Shock suppressors (snubbers) are used on piping systems or equipment to limit displacement from dynamic loads such as earthquake or thermal-hydraulic transient, while allowing displacement from thermal expansion. The consequences of an inoperable shock suppressor are a possible inoperable system due to increased loads on the associated piping or equipment from either the dynamic load or thermal expansion depending on the nature of the snubber problem. Shock suppressors are not active components like valves or pumps but are simply a type of support like springs, baseplates, or struts with the same potential for impact on OPERABILITY as any support. The majority of the snubbers at ANO-1 are installed on seismic class I lines which includes all of the safety systems.

Comparison to Screening Criteria:

Criterion 1

A snubber is not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

A snubber is not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

Snubbers are part of the support structure of a number of systems which are part of the primary success path and which function or actuate 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. However, the OPERABILITY of the snubbers, as with any support, is enveloped by the OPERABILITY of the system and is covered by LCO's applicable to the system. Therefore, an LCO for snubbers would be secondary and redundant to system LCO's.

Criterion 4

Requirements for snubber OPERABILITY are not specifically addressed in the ANO-1 PSA and are not credited in any accident analysis and are therefore determined to be non-risk significant with respect to core damage frequency and offsite releases.

Conclusion:

Snubbers satisfy the requirements of Criterion 3 but in a manner secondary to Technical Specifications for the systems of which they may be part. In this circumstance, if the snubber is declared inoperable then it is still possible, (and likely), that the system is OPERABLE and applicable system LCO's take precedence. Since the screening criteria for inoperable snubbers have either not been satisfied or shown to be redundant, the requirements may be relocated to other licensee controlled documents outside the Technical Specifications.

3.22 REACTOR BUILDING PURGE FILTRATION SYSTEM

LCO Statement:

3.22.1 The reactor building purge filtration system shall be operable whenever irradiated fuel handling operations are in progress in the reactor building and shall have the following performance capabilities:

LCO Related Requirements:

3.22 Applic, 3.22 Obj, 3.22.1 a, 3.22.1 b, 3.22.1 c, 3.22.1 d, 3.22.1 e, 3.22.2, 3.22.2 a, 3.22.2 b, 3.22.3, Bases & 4.25.

Discussion:

The reactor building purge filtration system is designed to filter the reactor building atmosphere during normal operations for ease of personnel entry into the reactor building. The system is required operable during fuel handling operations to limit the impact of a release of radioactive material should a fuel assembly be damaged. The system consists of a supply fan, a filter train, and an exhaust fan in series. The filter train consists of a pre-filter, a HEPA filter and a charcoal adsorber in series. A new analysis of the consequences of a fuel handling accident in the Reactor Building with the personnel air lock doors open has concluded that the 10CFR100 limits are met without the RB purge system operating.

Comparison to Screening Criteria:

Criterion 1

The reactor building purge filtration system is not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

The reactor building purge filtration system is not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

The reactor building purge filtration system is not 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.

Criterion 4

The reactor building purge filtration system is not addressed in the ANO-1 PSA and is not credited in an accident analysis and is therefore determined to be non-risk significant with respect to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the requirements for the reactor building purge filtration system may be relocated to other licensee controlled documents outside the Technical Specifications.

TABLE 4.1-1 INSTRUMENT SURVEILLANCE REQUIREMENTS

LCO Statement:

The following instrument channel Surveillance Requirements from Table 4.1-1 imply LCOs exist, however, unique LCOs are not specifically identified in Section 3 of the CTS:

Table 4.1-1 #38. Sodium Hydroxide Tank Level Indicator

LCO Related Requirements:

4.1 a, Table 4.1-1 NOTE, & Bases

Discussion:

Surveillance requirements shall be met during operational modes or other conditions specified for Limiting Conditions for Operation. Failure to perform a Surveillance Requirement within the allowed surveillance interval shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. CTS Table 4.1-1 lists instrument surveillance requirements which implies that there are corresponding LCOs for these items in CTS Section 3. A comparison of the table with the LCOs has resulted in one item to be relocated but which has an implied LCO. Furthermore, a comparison of these items with Regulatory Guide 1.97 also finds no match.

Comparison to Screening Criteria:

Criterion 1

Instrument surveillances for the sodium hydroxide tank level indicator are not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

Instrument surveillances for sodium hydroxide tank level indicator are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

Instrument surveillances for the sodium hydroxide tank level indicator are not 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.

Criterion 4

Instrument surveillances for the sodium hydroxide tank level indicator are not addressed in the ANO-1 PSA and are not credited in any accident analysis and are therefore determined to be non-risk significant with respect to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the requirements for instrument surveillances for the sodium hydroxide tank level indicator may be relocated to other licensee controlled documents outside the Technical Specifications.

TABLE 4.1-2 MINIMUM EQUIPMENT TEST FREQUENCY

LCO Statement:

The following minimum equipment test frequency requirements from Table 4.1-2 imply LCOs exist, however, unique LCOs are not specifically identified in Section 3 of the CTS:

Table 4.1-2 #5.	Refueling System Interlocks, and
Table 4.1-2 #10.	Spent Fuel Cooling System.

LCO Related Requirements:

4.1 b & Bases

Discussion:

Minimum surveillance requirements shall be met during operational modes or other conditions specified for Limiting Conditions for Operation. Failure to perform a Surveillance Requirement within the allowed surveillance interval shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. CTS Table 4.1-2 lists minimum equipment test frequency requirements which implies that there are corresponding LCOs for these items in CTS Section 3. A comparison of the table with the LCOs has resulted in the identification of these two items for relocation but for which there is only an implied LCO.

Comparison to Screening Criteria:

Criterion 1

Minimum equipment test frequencies for the refueling system interlocks and the spent fuel cooling system are not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

Minimum equipment test frequencies for the refueling system interlocks and the spent fuel cooling system are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

Minimum equipment test frequencies for the refueling system interlocks and the spent fuel cooling system are not 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.

Criterion 4

Minimum equipment test frequencies for the refueling system interlocks and the spent fuel cooling system are not addressed in the ANO-1 PSA and are not credited in any accident analysis and are therefore determined to be non-risk significant with respect to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the requirements for minimum equipment test frequencies for the refueling system interlocks and the spent fuel cooling system may be relocated to other licensee controlled documents outside the Technical Specifications.

TABLE 4.1-3 MINIMUM SAMPLING AND ANALYSIS FREQUENCY

__LCO Statement:__

The following minimum sampling and analysis frequencies from Table 4.1-3 imply LCOs exist, however, unique LCOs are not specifically identified in Section 3 of the CTS or the associated LCO is being relocated:

Table 4.1-3 #1.f.	Boron Concentration (for MODES 1, 2, 3, 4, and 5)
Table 4.1-3 #5.a.	Gross Radioiodine Concentration (with Notes 5, 7, and 10)

__LCO Related Requirements:__

4.1 b, Table 4.1-3 Notes 5, 7, & 10 (invoked for #5.a), & Bases

__Discussion:__

Minimum surveillance requirements shall be met during operational modes or other conditions specified for Limiting Conditions for Operation. Failure to perform a Surveillance Requirement within the allowed surveillance interval shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. CTS Table 4.1-3 lists minimum sampling and analysis frequency requirements which implies that there are corresponding LCOs for these items in CTS Section 3. A comparison of the table with the LCOs has resulted in two items being identified as not having a match and which will be relocated.

The Boron Concentration test implied LCO is relocated for performance during MODES 1, 2, 3, 4, and 5. This surveillance is retained in ITS 3.9 for MODE 6 and is associated with LCO 3.9.1 in SR 3.9.1.1.

In these tables, the Notes are invoked uniquely, as applicable, for each item. For the Gross Radioiodine Concentration test, the Notes invoked for this item are also relocated with the item. It should be noted that this does not relocate the Note for other test items which may have been retained.

__Comparison to Screening Criteria:__

__Criterion 1__

Minimum surveillance requirements for boron concentration in RCS samples in MODES 1, 2, 3, 4, and 5 and gross radioiodine concentration in secondary coolant samples do not constitute an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

Minimum surveillance requirements for boron concentration in RCS samples in MODES 1, 2, 3, 4, and 5 and gross radioiodine concentration in secondary coolant samples are not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier. Secondary coolant activity limits will be retained in ITS 3.7.4, "Secondary Specific Activity." These limits will preserve the initial conditions of the safety analysis. The boron concentration and gross radioiodine limits, however, do not reflect initial condition assumptions in the DBA.

Criterion 3

Minimum surveillance requirements for boron concentration in RCS samples in MODES 1, 2, 3, 4, and 5 and gross radioiodine concentration in secondary coolant samples are not 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.

Criterion 4

Minimum surveillance requirements for boron concentration in RCS samples in MODES 1, 2, 3, 4, and 5 and gross radioiodine concentration in secondary coolant samples are not addressed in the ANO-1 PSA and are not credited in any accident analysis and are therefore determined to be non-risk significant with respect to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the requirements for minimum surveillance requirements for boron concentration in RCS samples in MODES 1, 2, 3, 4, and 5 and gross radioiodine concentration in secondary coolant samples may be relocated to other licensee controlled documents outside the Technical Specifications.

4.6.3 EMERGENCY LIGHTING

LCO Statement:

The following Surveillance Requirement results in an implied LCO in that an LCO is not specifically identified in Section 3 of the CTS:

4.6.3 Emergency Lighting
The correct functioning of the emergency lighting system shall be verified once every 18 months.

LCO Related Requirements:

NA

Discussion:

Testing of the emergency lighting system is scheduled every 18 months but is subject to review and modification based on experience. The 18 month cycle is compatible with the period of simulated loss-of-power tests.

Comparison to Screening Criteria:

Criterion 1

Testing of the emergency lighting system is not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

Testing of the emergency lighting system is not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

Testing of the emergency lighting system is not 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.

Criterion 4

The need for lighting to perform operator actions in the Control Room and elsewhere during events is addressed in the ANO-1 PSA recovery actions. However, the testing of the emergency lighting system is not addressed. The testing of the emergency lighting system is therefore considered to be non-risk significant with respect to core damage frequency and offsite releases since the operators have access to flashlights for those recoveries in which lighting is necessary..

Conclusion:

Since the screening criteria have not been satisfied, the requirements for testing of the emergency lighting system may be relocated to other licensee controlled documents outside the Technical Specifications.

4.7.2 CONTROL ROD PROGRAM VERIFICATION (GROUP VS CORE POSITIONS)

LCO Statement:

The following Surveillance Requirements result in an implied LCO, in that an LCO is not specifically identified in Section 3 of the CTS:

4.7.2.1 Whenever the control rod drive patch panel is reconnected (after test, reprogramming, or maintenance), each control rod drive mechanism shall be selected from the control room and exercised by movement of sufficient travel to verify that the proper rod has responded as shown on the unit computer printout or on the input to the computer for that rod.

4.7.2.2 Whenever power or instrumentation cables to the control rod drive assemblies atop the reactor or at the bulkhead are disconnected or removed, an independent verification check of their reconnection shall be performed.

4.7.2.3 Any rod found to be improperly programmed shall be declared inoperable until properly programmed.

LCO Related Requirements:

4.7.2, 4.7.2 Applic, 4.7.2 Obj, & Bases

Discussion:

When test, reprogramming or maintenance of the control rod drive patch panel and associated cables and instrumentation is performed, the control rod control "programming" must be validated. Each control rod has a relative and an absolute position indicator system. One set of outputs goes to the plant computer identified by a unique number associated with only one core position. The other set of outputs goes to a programmable bank of 68 edgewise meters in the control room. In the event that a patching error is made in the patch panel or connectors in the cables leading to the control rod drive assemblies or the control room meter bank is improperly transposed upon reconnection, these errors and transpositions will be discovered by a comparative check. This type of comparative check, however, will not assure detection of improperly connected cables inside the reactor building. These cables require verification by an independent person who is cognizant of the proper configuration.

Comparison to Screening Criteria:

Criterion 1

Verification of the control rod program after test, reprogramming or maintenance is not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

Verification of the control rod program after test, reprogramming or maintenance is not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

Verification of the control rod program after test, reprogramming or maintenance is not 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.

Criterion 4

The ATWS portion of the PSA addresses the mechanical and electrical failures associated with the control rod drive mechanisms. Testing, programming, and maintenance are not considered in this model. ATWS has a low probability of failure associated with the rods ability to drop and scram the plant. The surveillances required in this specification are considered to be non-risk significant with respect to core damage frequency and offsite releases.

Conclusion:

Since the screening criteria have not been satisfied, the requirements for verification of the control rod program after test, reprogramming or maintenance may be relocated to other licensee controlled documents outside the Technical Specifications.

4.15 AUGMENTED INSERVICE INSPECTION PROGRAM FOR HIGH ENERGY LINES OUTSIDE OF CONTAINMENT

LCO Statement:

4.15.1 At the first refueling outage period, a volumetric examination shall be performed with 100 percent inspection of each weld in accordance with the requirements of ASME Code Section XI, Inservice Inspection of Nuclear Power Plant Components, to establish system integrity and baseline data.

4.15.2 The inservice inspection at each weld shall be performed in accordance with the requirements of ASME Code Section XI, Inservice Inspection of Nuclear Power Plant Components, with the following schedule:

(The inspection intervals identified below sequentially follow the baseline examination of 4.15.1).

First Inspection Interval

a. First 3-1/3 years (or nearest refueling outage)	100% volumetric inspection of each weld
b. Second 3-1/3 years (or nearest refueling outage)	100% volumetric inspection of each weld
c. Third 3-1/3 years (or nearest refueling outage)	100% volumetric inspection of each weld

Successive Inspection Intervals

Every 10 years thereafter (or nearest refueling outage)	Volumetric inspection of two of the welds at the expiration of each 1/3 of the inspection intervals with a cumulative 100% coverage of all welds.

Note - The welds selected during each inspection period shall be distributed among the total number to be examined to provide a representative sampling of the conditions of the welds.

LCO Related Requirements:

4.15 Applic, 4.15 Obj, 4.15.3, 4.15.4, & 4.15.5

Discussion:

There are six welds in the main steam and main feedwater lines located outside of the reactor building where protection from the consequences of postulated ruptures is not provided by a system of pipe whip restraints, jet impingement barriers, protective enclosures and/or other measures designed specifically to cope with such ruptures. These welds receive an augmented inspection which enhances the integrity of the pipe and reduces the probability of catastrophic failure. The inspection is performed in accordance with ASME Section XI and is a sequential volumetric inspection. Repairs, reexaminations and piping pressure tests, as required, are also performed in accordance with ASME Section XI.

Comparison to Screening Criteria:

Criterion 1

Augmented testing of six welds in the main steam and main feedwater lines located outside of the reactor building is not an instrumentation system that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

Augmented testing of six welds in the main steam and main feedwater lines located outside of the reactor building is not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or challenge to the integrity of a fission product barrier.

Criterion 3

Augmented testing of six welds in the main steam and main feedwater lines located outside of the reactor building is not 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.

Criterion 4

Piping integrity is considered in the PSA only in relation to initiating event frequency. For this LCO, the PSA model of a steamline/feedwater line break initiating event applies. A qualitative assessment assuming an increase in the initiating event frequency by a factor of two orders of magnitude reveals no significant increase in the overall core damage frequency.

Conclusion:

Since the screening criteria have not been satisfied, the requirements for augmented testing of six welds in the main steam and main feedwater lines located outside of the reactor building may be relocated to other licensee controlled documents outside the Technical Specifications.

Items on this page also addressed in the following packages: 3,4 A

SR

3.1.2 Pressurization, Heatup, and Cooldown Limitations

Specification

3.1.2.1 Hydro Tests

For thermal steady state system hydro tests, the system may be pressurized to the limits set forth in Specification 2.2 when there are fuel assemblies in the core, under the provisions of 3.1.2.3, and to ASME Code limits when no fuel assemblies are present provided the reactor coolant system limits are to the right of and below the limit line in Figure 3.1.2-1. The provisions of Specifications 3.0.3 are not applicable.

3.1.2.2 Leak Tests

Leak tests required by Specification 4.3 shall be conducted under the provision of 3.1.2.3. The provisions of Specification 3.0.3 are not applicable.

3.1.2.3 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1.2-2 and Figure 3.1.2-3, and are as follows:

Heatup:

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1.2-2. The heatup rates shall not exceed those shown in Figure 3.1.2-2.

Cooldown:

Allowable combinations of pressure and temperature for a specific cooldown shall be to the right of and below the limit line in Figure 3.1.2-3. Cooldown rates shall not exceed those shown in Figure 3.1.2-3.

~~3.1.2.4 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100F.~~ R TRM

~~3.1.2.5 The pressurizer heatup and cooldown rates shall not exceed 100F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430F.~~ R TRM

~~3.1.2.6 With the limits of Specifications 3.1.2.3 or 3.1.2.4 or 3.1.2.5 exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS Tavg to less than 200F, while maintaining RCS temperature and pressure below the curve, within the following 30 hours.~~ R TRM

Items on this page also addressed in the following packages: 3.4A, 3.4B

SR

BASES

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes.[1] These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation.[2] The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the DTT.[3]

R
TRM

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10CFR50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in FTI Document 77-1258569-01 [4]. The limiting weld material is being irradiated as part of the B&W Owners Group Integrated Reactor Vessel Material Surveillance Program and the identification and locations of the capsules containing the limiting weld material is discussed in the latest revision to B&W report, BAW-1543. [5] The chemical composition of the limiting weld material is reported in the B&W Report, BAW-2121P [6]. The effect of neutron irradiation on the RT_{NDT} of the limiting weld material is reported in FTI Calculations 32-1245917-00 and 32-1257716-00 [7].

Figures 3.1.2-1, 3.1.2-2, and 3.1.2-3 present the pressure-temperature limit curves for hydrostatic test, normal heatup, and normal cooldown respectively. The limit curves are applicable through the thirty first effective full power year of operation. The service period was reduced by one effective full power year from that assumed in FTI Document 77-1258569-01 to be conservative with respect to independent calculations performed by the NRC staff. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all allowed operating reactor coolant pump combinations.

The pressure-temperature limit lines shown on Figure 3.1.2-2 for reactor criticality and on Figure 3.1.2-1 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50 for reactor criticality and for inservice hydrostatic testing.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10CFR50, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core region.

The spray temperature difference restriction based on a stress analysis of the spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

R
TRM

Items on this page also addressed in the following packages: 3.4A, 3.4B

SR

The heatup and cooldown rates stated in this specification are intended as the maximum changes in temperature in one direction in a one hour period. The actual temperature linear ramp rate may exceed the stated limits for a time period provided that the maximum total temperature difference does not exceed the limit and that a temperature hold is observed to prevent the total temperature difference from exceeding the limit for the one hour period.

Specification 3.1.2.9 is to ensure that the core flood tanks are not the source for pressurizing the reactor coolant system when in cold shutdown.

Specification 3.1.2.10 is to ensure that high pressure injection is not the source of pressurizing the reactor coolant system when in cold shutdown. The LTOP enable temperature has been calculated in accordance with Code Case N-514. Instrument error is not included in the reactor coolant temperature of 262°F.

Specification 3.1.2.11 is to ensure that the reactor coolant system is not operated in a manner which would allow overpressurization due to a temperature transient.

<u>REFERENCES</u>

(1) ~~FSAR, Section 4.1.2.4~~

(2) ~~ASME Boiler and Pressure Code, Section III, N-415~~ R TRM

(3) ~~FSAR, Section 4.3.11.5~~

(4) FTI Document Number 77-1258569-01

(5) BAW-1543, latest revision

(6) BAW-2121P

(7) FTI Calculation Numbers 32-1245917-00 and 32-1257716-00

Items on this page also addressed in the following packages: 3.3A, 3.3B, 3.3C, 3.3D, 3.4B

SR

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objectives

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

3.5.1.1 Startup and operation are not permitted unless the requirements of Table 3.5.1-1, columns 3 and 4 are met.

3.5.1.2 In the event the number of protection channels operable falls below the limit given under Table 3.5.1-1, Columns 3 and 4, operation shall be limited as specified in Column 5.

3.5.1.3 For on-line testing or in the event of a protection instrument or channel failure, a key operated channel bypass switch associated with each reactor protection channel may be used to lock the channel trip relay in the untripped state as indicated by a light. Only one channel shall be locked in the untripped state or contain inoperable functions in the untripped state at any one time. In the event more than one protection channel contains inoperable functions in the untripped state, or a protection channel or function becomes inoperable concurrent with another protection channel locked in the untripped state, within 1 hour implement the actions required by Table 3.5.1-1 Note 6. Only one channel bypass key shall be accessible for use in the control room. While operating with an inoperable function unbypassed in the untripped state, the remaining RPS key operated channel bypass switches shall be tagged to prevent their operation.

3.5.1.4 The key operated shutdown bypass switch associated with each reactor protection channel shall not be used during reactor power operation except during channel testing. R TRM

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall be maintained in the source range until the one decade overlap is achieved.

3.5.1.6 In the event that one of the trip devices in either of the sources supplying power to the control rod drive mechanisms fails in the untripped state, the power supplied to the rod drive mechanisms through the failed trip device shall be manually removed within 30 minutes following detection. The condition will be corrected and the remaining trip devices shall be tested within eight hours following detection. If the condition is not corrected and the remaining trip devices are not tested within the eight-hour period, the reactor shall be placed in the hot shutdown condition within an additional four hours.

Items on this page also addressed in the following packages: 3.3A, 3.3B, 3.3C, 3.3D

SR

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless the requirements of Table 3.5.1-1, Columns 3 and 4, are met.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column 4 (Table 3.5.1-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR, Section 7.

There are four reactor protection channels. Normal trip logic is two-out-of-four. Required trip logic for the power range instrumentation channels is two-out-of-three. Minimum trip logic on other instrumentation channels is one-out-of-two.

The four reactor protection channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided with alarm and lights to indicate when that channel is bypassed. There will be one reactor protection system channel bypass switch key permitted in the control room. Upon the discovery of inoperable functions in any one reactor protection channel, the effect of the failure on the reactor protection system and other interconnected systems is evaluated. The affected reactor protection channel may be placed in channel bypass, remain in operation in a degraded condition, or placed in the tripped condition as determined by operating conditions and management judgment. This action allows placing the plant in the safest condition possible considering the extent of the failure, plant conditions, and guidance from plant management. Should the failure in the reactor protection channel prohibit the proper operation of another system, the appropriate actions for the affected system are implemented. Administrative controls are established to preclude placing a reactor protection channel in channel bypass when any other reactor protection channel contains an inoperable function in the untripped state.

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used.

R TRM

The source range and intermediate range nuclear flux instrumentation scales overlap by one decade. This decade overlap will be achieved at 10^{-10} amps on the intermediate range scale.

The ESAS employs three independent and identical analog channels, which supply trip signals to two independent, identical digital subsystems. In order to actuate the safeguards systems, two out of three analog channels must trip. This will cause both digital subsystems to trip. Tripping of either digital subsystem will actuate all safeguards systems associated with that digital subsystem.

Because only one digital subsystem is necessary to actuate the safeguards systems and these systems are capable of tripping even when they are being tested, a single failure in a digital subsystem cannot prevent protective action.

Items on this page also addressed in the following packages: 3.8, 3.9

SR

3.8 FUEL LOADING AND REFUELING

Applicability

Applies to fuel loading and refueling operations.

Objective

To assure that fuel loading, refueling and fuel handling operations are performed in a responsible manner.

Specification

R TRM

3.8.1 ~~Radiation levels in the reactor building refueling area shall be monitored by instrument RE-8017. Radiation levels in the spent fuel storage area shall be monitored by instrument RE-8009. If any of these instruments become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.~~

3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, each with continuous indication available, whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

3.8.3.a. At least one decay heat removal loop shall be in operation.* Otherwise, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the reactor coolant system, and close all containment penetrations providing access from the containment atmosphere to the outside atmosphere within 4 hours.

b. When the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet, two decay heat removal loops shall be operable.**

Otherwise, immediately initiate corrective action to return the required loops to operable status as soon as possible.

3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required for refueling shutdown.

R TRM

3.8.5 ~~Direct communications between the control room and the refueling personnel in the reactor building shall exist whenever changes in core geometry are taking place.~~

*The decay heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of core alterations.

**The normal or emergency power source may be inoperable for each shutdown cooling loop.

SR

Items on this page are also addressed in package 3.9

3.8.6 During the handling of irradiated fuel in the reactor building, at least one door on the personnel and emergency hatches shall be capable* of being closed. The equipment hatch cover shall also be capable* of being closed. At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

3.8.7 Isolation valves in lines containing automatic containment isolation valves shall be operable, or at least one shall be closed.

3.8.8 ~~When two irradiated fuel assemblies are being moved simultaneously by the bridges within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times.~~ R TRM

3.8.9 ~~If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made. The provisions of Specification 3.0.3 are not applicable.~~ R TRM

3.8.10 The reactor building purge isolation system, including the radiation monitors shall be tested and verified to be operable within 7 days prior to refueling operations. The provisions of Specification 3.0.3 are not applicable.

3.8.11 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 100 hours. In the event of a complete core offload, a full core to be discharged shall be subcritical a minimum of 175 hours prior to discharge of more than 70 assemblies to the spent fuel pool. The provisions of Specification 3.0.3 are not applicable.

3.8.12 ~~All fuel handling in the Auxiliary Building shall cease upon notification of the issuance of a tornado watch for Pope, Yell, Johnson, or Logan counties in Arkansas. Fuel handling operations in progress will be completed to the extent necessary to place the fuel handling bridge and crane in their normal parked and locked position. The provisions of Specification 3.0.3 are not applicable.~~ R TRM

3.8.13 ~~No loaded spent fuel shipping cask shall be carried above or into the Auxiliary Building equipment shaft unless atmospheric dispersion conditions are equal to or better than those produced by Pasquill Type D stability accompanied by a wind velocity of 2 m/sec. In addition, the railroad spur door of the Turbine Building shall be closed and the fuel handling area ventilation system shall be in operation. The provisions of Specification 3.0.3 are not applicable.~~ R TRM

3.8.14 ~~Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool. The provisions of Specification 3.0.3 are not applicable.~~ R TRM

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* Administrative controls shall ensure that appropriate personnel are aware that both personnel airlock doors and/or equipment hatch are open, a specific individual(s) is designated and available to close an airlock door and the equipment hatch cover following a required evacuation of the reactor building, and any obstruction(s) (e.g., cables and hoses) that could prevent closure of an airlock door and the equipment hatch cover be capable of being quickly removed.

Items on this page are also addressed in packages 3.7 and 3.9

SR

replacement. The keff with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing reactor building purge termination is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

~~Because of physical dimensions of the fuel bridges, it is physically impossible for fuel assemblies to be within 10 feet of each other while being handled.~~ R TRM

Per specification 3.8.6, the reactor building personnel and/or emergency airlock doors and the equipment hatch may be open during movement of irradiated fuel in the reactor building provided at least one door of each airlock and the equipment hatch are capable of being closed in the event of a fuel handling accident and the plant is in REFUELING SHUTDOWN with 23 feet of water above the fuel seated within the reactor pressure vessel. Should a fuel handling accident occur inside the reactor building, at least one of the personnel and/or emergency airlock doors and the equipment hatch will be closed following evacuation of the reactor building. For closure, the equipment hatch cover will be in place with a minimum of four bolts securing the cover to the sealing surface.

Specification 3.8.11 is required as: 1) the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 100 hours (3); and, 2) to assure that the maximum design heat load of the spent fuel pool cooling system will not be exceeded during a full core offload.

~~Specification 3.8.14 will assure that damage to fuel in the spent fuel pool will not be caused by dropping heavy objects onto the fuel. Administrative controls will prohibit the storage of fuel in locations adjoining the walls at the north and south ends of the pool, in the vicinity of cask storage area and fuel tilt pool access gates.~~ R TRM

Specifications 3.8.15 and 3.8.16 assure fuel enrichment and fuel burnup limits assumed in the spent fuel safety analyses will not be exceeded.

Specification 3.8.17 assures the boron concentration in the spent fuel pool will remain within the limits of the spent fuel pool accident and criticality analyses.

REFERENCES

(1) FSAR, Section 9.5

(2) FSAR, Section 14.2.2.3

(3) ~~FSAR, Section 14.2.2.3.3~~ R TRM

Items on this page also addressed in the following packages: NA

SR

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TRM

3.16 Shock Suppressors (Snubbers)

Applicability

This technical specification applies to all shock suppressors (snubbers). The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

Objective

To assure adequate shock suppression protection for primary coolant system piping and any other safety related system or component under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. This is done by assuring the operability of those shock suppressors installed for that purpose.

Specification

3.16.1 With one or more applicable snubbers inoperable, within 72 hours either:

a. Replace or restore the inoperable snubbers to an OPERABLE status and perform an engineering evaluation of the attached components per Specification 4.16.1.f or,

b. Perform a review and evaluation which justifies continued operation with the inoperable snubber(s) and perform an engineering evaluation of the attached component(s) per Specification 4.16.1.f or,

c. Declare the attached system inoperable and follow the appropriate ACTION statement for that system.

Bases

Shock suppressors are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable shock suppressor is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all shock suppressors required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the shock suppressor protection is required only during low probability events, a period of 72 hours is allowed for repairs, replacements or evaluations. If a reveiw and evaluation of an INOPERABLE snubber is preformed and documented to justify continued operation, and provided all design criteria are met with the INOPERABLE snubber, then the INOPERABLE snubber would not need to be restored or replaced. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures.

Items on this page also addressed in the following packages: NA

SR

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TRM

3.22 REACTOR BUILDING PURGE FILTRATION SYSTEM

Applicability

This specification applies to the operability of the reactor building purge filtration system.

Objective

To assure that the reactor building purge filtration system will perform within acceptable levels of efficiency and reliability.

Specification

3.22.1 The reactor building purge filtration system shall be operable whenever irradiated fuel handling operations are in progress in the reactor building and shall have the following performance capabilities:

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows (±10%) on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal at a velocity within ±20% of system design, 0.05 to 0.15 mg/m^3 inlet methyl iodide concentration, ≥70% R. H. and ≥125F.

c. Fans shall be shown to operate within ±10% design flow.

d. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate (±10%).

e. Air distribution shall be uniform within ±20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the reactor building purge filtration system.

3.22.2 If the requirements of Specification 3.22.1 cannot be met, either:

a. Irradiated fuel movement shall not be started (any irradiated fuel assembly movement in progress may be completed); or,

b. Isolate the reactor building purge system.

3.22.3 The provisions of Specification 3.0.3 are not applicable.

Bases

The reactor building purge filtration system is designed to filter the reactor building atmosphere during normal operations for ease of personnel entry into the reactor building. This specification is intended to require the system operable during fuel handling operations, if the system

Items on this page also addressed in the following packages: NA SR

R TRK

is to be used, to limit the release of activity should a fuel handling accident occur. The system consists of one circuit containing a supply and an exhaust fan and a filter train. The filter train consists of a pre-filter, a HEPA filter and a charcoal adsorber in series.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

Items on this page also addressed in the following packages: 3.3A, 3.3B, 3.3C, 3.3D, 5.5

SR

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requiremetns of any Technical Specification.

4.1 OPERATIONAL SAFETY ITEMS

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

R TRM

a. ~~The minimum frequency and type of surveillance required for reactor protective system and engineered safeguards system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.~~

Items on this page also addressed in the following packages: 3.0, 3.2, 3.3A, 3.3B, 3.3C, 3.3D, 5.5

SR

OPERATIONAL SAFETY ITEMS (continued)

4.1 (Continued)

b. Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3.

R

TRM

c. Discrepancies noted during surveillance testing will be corrected and recorded.

d. A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system.

BASES

4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):

"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.

Items on this page also addressed in the following packages: 3.3A, 3.3B, 3.3C, 3.3D, 5.5

SR

BASES (continued)

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an operational mode or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

4.1 Bases

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator Action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

R TRM

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels shall be calibrated at least twice weekly (during steady state operating conditions) against a heat balance standard to compensate for instrumentation drift. During nonsteady state operation, the nuclear flux channels shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters.

Items on this page also addressed in the following packages: 3.3 A, 3.3 B, 3.3 C, 3.3 D

SR

R
TRM

~~The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the status of the equipment and systems to assure safe operation.~~

~~**REFERENCE**~~

~~FSAR Section 7.1.2.3.4~~

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
36. Boric Acid Addition Tank				
a. Level Channel	NA	NA	R	
b. Temperature Channel	M	NA	R	
37. Degraded Voltage Monitoring	W	R	R	
38. Sodium Hydroxide Tank Level Indicator	NA	NA	R	R TRM
39. Incore Neutron Detectors	M(1)	NA	NA	(1) Check Functioning
40. Emergency Plant Radiation Instruments	M(1)	NA	R	(1) Battery Check
41. Reactor Trip Upon Turbine Trip Circuitry	M	PC	R	
42. Deleted				

Items on this page also addressed in the following packages: 3.2, 3.3A, 3.3D, 3.5

SR

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
d. SG A High Range Level High-high	S	M	R	
e. SG B High Range Level High-high	S	M	R	
57. Containment High Range Radiation Monitors	D	M	R	
58. Containment Pressure-High	M	NA	R	
59. Containment Water Level-Wide Range	M	NA	R	
60. Low Temperature Overpressure Protection Alarm Logic	NA	R	R	
61. Core-exit Thermocouples	M	NA	R	
62 Electronic (SCR) Trip Relays	NA	Q	NA	
63 RVLMS	M	NA	R	
64 HLLMS	M	NA	R	

NOTE:

S - Each Shift
W - Weekly
M - Monthly
D - Daily

T/W - Twice per Week
Q - Quarterly
P - Prior to each startup if not done previous week
B/M - Every 2 months

R - Once every 18 months
PC - Prior to going critical if not done within previous 31 days
NA - Not Applicable
SA - SA Twice per Year

(R) TRM

Items on this page are also addressed in packages 3.3A 3.3B 3.3C 3.3D 3.4B

SR

Items on this page also addressed in the following packages: 3.1, 3.4B, 3.6, 3.7 TR

Table 4.1-2
Minimum Equipment Test Frequency

Item	Test	Frequency
1. Control Rods	Rod Drop Times of all Full Length Rods 1/	Each Refueling Shutdown
2. Control Rod Movement	Movement of Each Rod	Every Two Weeks Above Cold Shutdown Conditions
3. Pressurizer Code Safety Valves	Setpoint	One Valve Every 18 Months
4. Main Steam Safety Valves	Setpoint	Four Valves Every 18 Months
~~5. Refueling System Interlocks~~	~~Functioning~~	~~Start of Each Refueling Shutdown~~ R TRM
6a. Reactor Coolant System Leakage	Evaluate	Daily
b. Reactor Coolant System Pressure Isolation Valves	Leakage Test Per Table 3.1.6.9	See Notes 1 & 2
7. Emergency-powered Pressurizer Heaters	Power availability	Daily
	Heater capacity functional test	Every 18 Months
8. Reactor Building Isolation Trip	Functioning	Every 18 Months
9. Service Water Systems	Functioning	Every 18 Months
~~10. Spent Fuel Cooling System~~	~~Functioning~~	~~Every 18 Months when irradiated fuel is in the pool~~ R TRM

1/ Same as tests listed in Section 4.7

Notes:

(1) Leak testing for each valve shall be individually accomplished to demonstrate operability following each refueling, following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement.

(2) Whenever integrity of a pressure isolation valve listed in Table 3.1.6.9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.

Items on this page also addressed in the following packages: 3.1, 3.4A, 3.4B, 3.5, 3.6, 3.7, 3.9

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Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency
1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)
	b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(7)
	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)
	d. Dissolved Gases	d. Weekly (7)
	e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)
	~~f. Boron Concentration~~	~~f. 3 times/week~~ R TRM
	g. Radiochemical Analysis for $\bar{E}$ Determination (2) (4)	g. Monthly (7)
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup
3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)
~~5. Secondary Coolant Samples~~	~~a. Gross Radioiodine Concentration~~	~~a. Weekly (5)(7)(10)~~ R TRM
	b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)
6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of µCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever the gross radioactivity concentration exceeds 10% of the limit specified in the Specification 3.1.4.1 or increases by 10 µCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

Items on this page also addressed in the following packages: 3.1, 3.4A, 3.4B, 3.7

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(2) A radiochemical analysis shall consist of the quantitative measurement of the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes shall be used in the determination of Ē. A radiochemical analysis and calculation of Ē and iodine isotopic activity shall be performed if the measured gross activity changes by more than 10 μCi/gm from the previous measured level. The gamma energy per disintegration for those radioisotopes determined to be present shall be as given in "Table of Isotopes" (1967) and beta energy per disintegration shall be as given in USNRDL-TR-802 (Part II) or other references using the equivalent values for the radioisotopes.

(3) In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 μCi/gm from the previous measured level.

(4) Iodine isotopic activities shall be weighted to give I-131 dose equivalent activity.

(5) In addition to the weekly measurement, the radioiodine concentration shall be determined if there are indications that the primary to secondary coolant leakage rate has increased by a factor of 2. R TRM

(6) Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 1 percent but less than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken within 24 hours of any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by the above.

Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken prior to any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by above.

(7) Not required when plant is in the cold shutdown condition or refueling shutdown condition. R TRM

(8) O_2 analysis is not required when plant is in the cold shutdown condition or refueling shutdown condition.

(9) Required only when fuel is in the pool and prior to transferring fuel to the pool.

(10) Not required when not generating steam in the steam generators. R TRM

(11) The following shall be required until the end of Cycle 2 operation:

a. Gross radioiodine shall be determined at least three times per week during power operation.

e. Diesel fuel from the emergency storage tank shall be sampled and found to be within acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water, and sediment.

5. Once every 31 days the pressure in the required starting air receiver tanks shall be verified to be ≥ 175 psig.

Once every 18 months, the capacity of each diesel oil transfer pump shall be verified to be at least 10 gpm.

4.6.2 DC Sources and Battery Cell Parameters

1. Verify battery terminal voltage is ≥ 124.7 V on float charge once each 7 days.

2. Verify battery capacity is adequate to supply, and maintain in operable status, the required emergency loads for the design duty cycle when subjected to either a battery service test or a modified performance discharge test once every 18 months.

3. Verify battery capacity is ≥ 80% of the manufacturers rating when subjected to a performance discharge test or a modified performance discharge test once every 60 months, once every 24 months when battery has reached 85% of the service life with capacity ≥ 100% of the manufacturers rating and showing no degradation, and once every 12 months when battery shows degradation or has reached 85% of the service life and capacity is < 100% of the manufacturer's rating.

4. Any battery charger which has not been loaded while connected to its 125V d-c distribution system for at least 30 minutes during every quarter shall be tested and loaded while connected to its bus for 30 minutes.

5. Verify battery pilot cell parameters meet Table 4.6-1 Category A limits once per 7 days.

6. Verify average electrolyte temperature of representative cells is ≥ 60°F once per 92 days.

7. Verify battery cell parameters meet Table 4.6-1 Category B limits once per 92 days and once within 24 hours after a battery discharge to < 110 V and once within 24 hours after a battery overcharge to > 145 V.

8. Verify electrolyte temperature of pilot cell is ≥ 60°F once per 31 days.

4.6.3 Emergency Lighting

The correct functioning of the emergency lighting system shall be verified once every 18 months.

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Items on this page are also addressed in package 3.8

Footnotes (b) and (c) to Table 4.6-1 are applicable to Category A, B, and C specific gravity. Footnote (b) to Table 4.6-1 requires the above mentioned correction for electrolyte temperature. The value of 2 amps used in footnote (c) is the nominal value for float current established by the battery vendor as representing a fully charged battery with an allowance for overall battery condition. This current provides, in general, an indication of overall battery condition.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450. Footnote (c) to Table 4.6-1 allows the float charge current to be used as an alternate to specific gravity for up to 7 days following a battery recharge. Within 7 days each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than 7 days.

The SR 4.6.3 testing of the emergency lighting is scheduled every 18 months and is subject to review and modification if experience demonstrates a more effective test schedule.

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<u>REFERENCE</u>

FSAR, Section 8

Items on this page also addressed in the following packages: NA

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R SAR

4.7.2 Control Rod Program Verification
(Group Vs Core Positions)

Applicability

Applies to surveillance of the control rod systems.

Objective

To verify that the designated control rod (by core position) is operating in its programmed functional position and group (rods 1 through 12, group 1-8).

Specification

4.7.2.1 Whenever the control rod drive patch panel is reconnected (after test, reprogramming, or maintenance), each control rod drive mechanism shall be selected from the control room and exercised by movement of sufficient travel to verify that the proper rod has responded as shown on the unit computer printout or on the input to the computer for that rod.

4.7.2.2 Whenever power or instrumentation cables to the control rod drive assemblies atop the reactor or at the bulkhead are disconnected or removed, an independent verification check of their reconnection shall be performed.

4.7.2.3 Any rod found to be improperly programmed shall be declared inoperable until properly programmed.

Bases

Each control rod has a relative and an absolute position indicator system. One set of outputs goes to the plant computer identified by a unique number associated with only one core position. The other set of outputs goes to a programmable bank of 68 edgewise meters in the control room. In the event that a patching error is made in the patch panel or connectors in the cables leading to the control rod drive assemblies or the control room meter bank is improperly transposed upon reconnection, these errors and transpositions will be discovered by a comparative check by (1) selecting a specific rod from one group (e.g., rod 1 in regulating group 6), (2) noting the program-approved core position for this rod of the group, (3) exercising the selected rod, and (4) noting that a) the computer prints out both absolute and relative position response for the approved core position, and b) the proper meter in the control room display bank indicates both absolute and relative meter positions. This type of comparative check will not assure detection of improperly connected cables inside the reactor building. For these, (Specification 4.7.2.2) it will be necessary for a responsible person, other than the one doing the work, to verify by appropriate means that each cable has been matched to the proper control rod drive assembly.

Items on this page also addressed in the following packages: NA SR

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4.15 AUGMENTED INSERVICE INSPECTION PROGRAM FOR HIGH ENERGY LINES OUTSIDE OF CONTAINMENT

Applicability

Applies to welds in piping systems located outside of containment where protection from the consequences of postulated ruptures is not provided by a system of pipe whip restraints, jet impingement barriers, protective enclosures and/or other measures designed specifically to cope with such ruptures.

For Arkansas Nuclear One-Unit 1 this specification applies to six welds in the main steam and main feedwater lines identified as welds 6, 7, 23, 24, 55, and 56 on Figures A-7, A-8 and A-15 of the Final Safety Analysis Report.

Objective

To provide assurance of the continued integrity of the piping system over their service lifetime.

Specifications

4.15.1 At the first refueling outage period, a volumetric examination shall be performed with 100 percent inspection of each weld in accordance with the requirements of ASME Code Section XI, Inservice Inspection of Nuclear Power Plant Components, to establish system integrity and baseline data.

4.15.2 The inservice inspection at each weld shall be performed in accordance with the requirements of ASME Code Section XI, Inservice Inspection of Nuclear Power Plant Components, with the following schedule:

(The inspection intervals identified below sequentially follow the baseline examination of 4.15.1).

First Inspection Interval

a. First 3-1/3 years (or nearest refueling outage)	100% volumetric inspection of each weld
b. Second 3-1/3 years (or nearest refueling outage)	100% volumetric inspection of each weld
c. Third 3-1/3 years (or nearest refueling outage)	100% volumetric inspection of each weld

Items on this page also addressed in the following packages: NA SR

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Successive Inspection Intervals

Every 10 years thereafter (or nearest refueling outage)	Volumetric inspection of two of the welds at the expiration of each 1/3 of the inspection intervals with a cumulative 100% coverage of all welds.

Note - The welds selected during each inspection period shall be distributed among the total number to be examined to provide a representative sampling of the conditions of the welds.

4.15.3 In the event repairs of any welds are required following any examination during successive inspection intervals, the inspection schedule for the repaired welds will revert back to the first 10 year inspection program.

4.15.4 Examinations that reveal unacceptable structural defects in a weld during an inspection under 4.15.2 should be extended to require an additional inspection of another 1/3 of the welds. If further unacceptable defects are detected in the second sampling, the remainder of the welds shall be inspected.

4.15.5 Repairs, reexamination and piping pressure tests shall be conducted in accordance with Section XI of the ASME Code.

Items on this page also addressed in the following packages: NA

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4.16 SHOCK SUPPRESSORS (Snubbers)

Applicability

This technical specification applies to all shock suppressors (snubbers). The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

Objective

Verify an acceptable level of operability of the shock suppressors protecting the primary system and any other safety related system or component.

Specification

4.16.1 The following surveillance requirements apply to all applicable shock suppressors.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers may be categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.16-1. The visual inspection interval for each category of snubber shall be determined based upon the criteria provided in Table 4.16.1.

Items on this page also addressed in the following packages: NA

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c. <u>Visual Inspection Acceptance Criteria</u>

Visual inspections shall verify (1) that there are no visible indications of damage or impaired operability, and (2) attachments to the foundation or supporting structure are functional and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as INOPERABLE and may be reclassified OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined operable per Specifications 4.16.1.d or 4.16.1.e, as applicable. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined operable via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to a common hydraulic fluid reservoir shall be evaluated for operability if any snubber connected to that reservoir is determined to be inoperable.

d. <u>Functional Tests</u>

At least once each refueling shutdown a representative sample of snubbers shall be tested using the following sample plan.

At least 10% of the snubbers required by Specification 3.16.1 shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.16.1.e, an additional 10% of the snubbers shall be functionally tested until no more failures are found or until all snubbers have been functionally tested.

The representative samples for the functional test sample plans shall be randomly selected from the snubbers required by Specification 3.16.1 and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of sizes, and capacities. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

Items on this page also addressed in the following packages: NA

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e. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression. except that inertia dependent, acceleration limiting mechanical snubbers may be tested to verify only that activation takes place in both directions of travel;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both direction of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

f. Functional Test Failure Analysis

An evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used. if applicable, in selecting snubbers to be tested in an effort to determine the operability of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to activate or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or

Items on this page also addressed in the following packages: NA

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design deficiency, all snubbers of the same type subject to the same defect shall be evaluated in a manner to ensure their operability. This testing requirement shall be independent of the requirements stated in Specification 4.16.1.d for snubbers not meeting the functional test acceptance criteria.

g. Preservice Testing of Repaired, Replacement and New Snubbers

Preservice operability testing shall be performed on repaired, replacement or new snubbers prior to installation. Testing may be at the manufacturer's facility. The testing shall verify the functional test acceptance criteria in Specification 4.16.1.e.

In addition, a preservice inspection shall be performed on each repaired, replacement or new snubber and shall verify that:

1) There are no visible signs of damage or impaired operability as a result of storage, handling or installation;

2) The snubber load rating, location, orientation, position setting and configuration (attachments, extensions, etc.), are in accordance with design;

3) Adequate swing clearance is provided to allow snubber movement;

4) If applicable, fluid is at the recommended level and fluid is not leaking from the snubber system;

5) Structural connections such as pins, bearings, studs fasteners and other connecting hardware such as lock nuts, tabs, wire and cotter pins are installed correctly.

h. Snubber Seal Replacement Program

The seal service life of hydraulic snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The expected service life for the various seals, seal materials, and applications shall be determined and established based on engineering information and the seals shall be replaced so that the expected service life will not be exceeded during a period when the snubber is required to be operable. The seal replacements shall be documented and the documentation shall be retained in accordance with Specification 6.9.2.

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TABLE 4.16-1
SNUBBER VISUAL INSPECTION INTERVAL

NUMBER OF INOPERABLE SNUBBERS

Population per Category (Notes 1 and 2)	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber category shall be determined based upon the previous inspection interval and the number of INOPERABLE snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, categories must be determined and documented before any inspection and that determination shall be the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population per category and the number of INOPERABLE snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, and C if that integer includes a fractional value of INOPERABLE snubbers as determined by interpolation.

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TABLE 4.16-1 (Continued)

SNUBBER VISUAL INSPECTION INTERVAL

Note 3: If the number of INOPERABLE snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of INOPERABLE snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of INOPERABLE snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of INOPERABLE snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of INOPERABLE snubbers found during the previous interval and the number in Column B to the difference in the numbers in Column B and C.

Note 6: Specified surveillance intervals may be adjusted plus or minus 25 percent to accommodate normal test and surveillance schedule intervals up to and including 48 months, with the exception that inspection of inaccessible snubbers may be deferred to the next shutdown when plant conditions allow five days for inspection. See Note 7 for definition of interval as applied to snubber visual inspections. The provisions of Specification 4 regarding surveillance intervals are not applicable.

Note 7: Interval as defined for the shock suppressors (snubbers) visual inspection surveillance requirements is the period of time starting when the unit went into cold shutdown for refueling, and ending when the unit goes into cold shutdown for its next scheduled refueling. This period of time is nominally considered to be an 18 month period, or a 24 month period based on the type of fuel being used. However, the period of time (interval) could be shorter or longer due to plant operating variables such as fuel life and operating performance.

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BASES

All safety related snubbers are required to be operable to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure, or failure of the system on which they are installed, would have no adverse effect on any safety related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to plant systems. Therefore, the required inspection interval varies based upon the number of INOPERABLE snubbers found during the previous inspection in proportion to the sizes of the various snubber populations or categories and the previous inspection interval as specified in NRC Generic Letter 90-09, "Alternative Requirements For Snubber Visual Inspection Intervals and Corrective Actions". Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the result of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety related component or system has been adversely affected by inoperability of the snubber. The engineering evaluation is performed to determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

If a review and evaluation of an INOPERABLE snubber is performed and documented to justify continued operation, and provided that all design criteria are met with the INOPERABLE snubber, then the INOPERABLE snubber would not need to be restored or replaced.

Items on this page also addressed in the following packages: NA

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4.25 Reactor Building Purge Filtration System

Applicability

Applies to the surveillance of the reactor building purge filtration system.

Objective

To verify an acceptable level of efficiency and operability of the reactor building purge filtration system.

Specification

4.25.1 The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (±10%) within 720 system operating hours prior to initial irradiated fuel handling operations.

4.25.2 Initially and after any maintenance or testing that could affect the air distribution within the reactor building purge system, air distribution shall be demonstrated to be uniform within ±20% across HEPA filters and charcoal adsorbers.

4.25.3a. The tests and sample analysis of Specification 3.22.1.a, b, & c. shall be performed within 720 system operating hours prior to initial irradiated fuel handling operations in the reactor building, and prior to irradiated fuel handling in the reactor building following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall also be performed prior to irradiated fuel handling in the reactor building after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall also be performed prior to irradiated fuel handling in the reactor building after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing.

Bases

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per refueling period to show system performance capability.

Items on this page also addressed in the following packages: NA

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The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant and of the HEPA filter bank with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d. of Regulatory Guide 1.52. Radioactive methyl iodide removal efficiency tests shall be performed in accordance with RDT Standard M16-IT. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbents qualified according to Regulatory Guide 1.52.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"R" - Relocation of requirements:

Relocating requirements which do not meet the Technical Specification selection criteria to documents with an established control program allows the Technical Specifications to be reserved only for those conditions or limitations upon reactor operation which are necessary to adequately limit the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, thereby focusing the scope of Technical Specifications.

Therefore, requirements which do not meet the Technical Specification selection criteria in 10 CFR 50.36 have been relocated to other controlled license basis documents. This regulation addresses the scope and purpose of Technical Specifications. In doing so, it establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. These criteria are as follows:

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

Criterion 2: A process variable that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

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 barrier.

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

The application of these criteria is provided in the "Application of Selection Criteria to the ANO-1 Technical Specifications." Requirements which met the criteria have been included in the proposed improved Technical Specifications. Entergy Operations proposes to remove the requirements which do not meet the criteria from the Technical Specifications and relocate the requirements to a suitable owner controlled document. The requirements in the relocated Specifications will not be affected by this Technical Specification change. Entergy Operations will initially continue to perform the required operation and maintenance to assure that the requirements are satisfied. Relocating specific requirements for systems or variables will have no impact on the system's operability or the variable's maintenance, as applicable.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

License basis document control mechanisms, such as 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls," will be utilized for the relocated Specifications as they will be placed in other controlled license basis documents. This would allow Entergy Operations to make changes to these requirements, without NRC approval, as allowed by the applicable regulatory requirements. These controls are considered adequate for assuring structures, systems and components in the relocated Specifications are maintained operable and variables in the relocated Specifications are maintained within limits.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

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

   The proposed change relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the ANO-1 Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled license basis document and maintained pursuant to the applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled license basis document for which future changes will be evaluated pursuant to the requirements of the applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"A" - Administrative changes to requirements:

Reformatting and rewording the remaining requirements in accordance with the style of the improved Babcock & Wilcox Standard Technical Specifications in NUREG-1430 will make the Technical Specifications more readily understandable to plant operators and other users. Application of the format and style will also assure consistency is achieved between specifications. As a result, the reformatting and rewording of the Technical Specifications has been performed to make them more readily understandable by plant operators and other users. During this reformatting and rewording process, no technical changes (either actual or interpretational) to the Technical Specifications were made unless they were identified and justified.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

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

   The proposed change involves reformatting and rewording of the existing Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, there is no technical change to the requirements and therefore, there is no significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"LA" - Less restrictive, Administrative deletion of requirements:

Portions of some Specifications provide information that is descriptive in nature regarding the equipment, system(s), actions or surveillances. This information is proposed to be deleted from the specification and relocated to other license basis documents which are under licensee control. These documents include the TS Bases, Safety Analysis Report (SAR), Technical Requirements Manual, and Programs and Manuals identified in ITS Section 5, "Administrative Controls." The removal of descriptive information is permissible, because the documents containing the relocated information will be controlled through the applicable process provided by the regulatory requirements, e.g., 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls." This will not impact the actual requirements but may provide some flexibility in how the requirement is conducted. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

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

   The proposed change relocates requirements from the Technical Specifications to other license basis documents which are under licensee control. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the Technical Specifications to other license basis documents, which are under licensee control, are the same as the existing Technical Specifications. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"M" - More restrictive changes to requirements:

The ANO-1 Technical Specifications are proposed to be modified in some areas to impose more stringent requirements than previously identified. These more restrictive modifications are being imposed to be consistent with the improved Babcock & Wilcox Standard Technical Specifications. Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

The modification of the ANO-1 Technical Specifications and the changes made to achieve consistency within the specifications have been performed in a manner such that the most stringent requirements are imposed, except in cases which are individually evaluated.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

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

   The proposed change provides more stringent requirements for the ANO-1 Technical Specifications. These more stringent requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not impact the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated for ANO-1.

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

The imposition of more stringent requirements prevents a reduction in the margin of plant safety by:

a) Increasing the analytical or safety limit,
b) Increasing the scope of the specification to include additional plant equipment,
c) Increasing the applicability of the specification,
d) Providing additional actions,
e) Decreasing restoration times,
f) Imposing new surveillances, or
g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

This Section Addresses the Following Specifications:

NUREG-1430	ANO-1 ITS	Title
1.1	1.1	Definitions
1.2	1.2	Logical Connectors
1.3	1.3	Completion Times
1.4	1.4	Frequency

1.0 USE AND APPLICATION

1.1 Definitions

---NOTE---
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ALLOWABLE THERMAL POWER	ALLOWABLE THERMAL POWER shall be the maximum steady state reactor core heat transfer rate to the reactor coolant permitted by consideration of the number and configuration of reactor coolant pumps (RCPs) in operation.
AXIAL POWER IMBALANCE	AXIAL POWER IMBALANCE shall be the power in the top half of the core, expressed as a percentage of RATED THERMAL POWER (RTP), minus the power in the bottom half of the core, expressed as a percentage of RTP.
AXIAL POWER SHAPING RODS (APSRs)	APSRs shall be the part-length control components used to control the axial power distribution of the reactor core. The APSRs are positioned manually by the operator and are not trippable.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel.

(continued)

1.1 Definition

CHANNEL CALIBRATION (continued)	The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total steps.
CONTROL RODS	CONTROL RODS shall be all full length safety and regulating rods that are used to shutdown the reactor and control power level during maneuvering operations.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the ANO-1 specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites."

(continued)

1.1 Definition (continued)

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

LEAKAGE

LEAKAGE shall be:

a. <u>Identified LEAKAGE</u>

1. LEAKAGE, such as that from pump seals or valve packing (except RCP seal water leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System;

b. <u>Unidentified LEAKAGE</u>

All LEAKAGE (except RCP seal water leakoff) that is not identified LEAKAGE;

c. <u>Pressure Boundary LEAKAGE</u>

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

(continued)

1.1 Definition (continued)

OPERABLE-OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in the SAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT (QPT)	QPT shall be defined by the following equation and is expressed as a percentage. $QPT = 100\left(\frac{\text{Power in any Core Quadrant}}{\text{Average Power in all Quadrants}} - 1\right)$
RATED THERMAL POWER (RTP)	RTP shall be a total steady state reactor core heat transfer rate to the reactor coolant of 2568 MWt.

(continued)

1.1 Definition (continued)

SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM; b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level; and c. There is no change in APSR position.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER [a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 280
4	Hot Shutdown [b]	< 0.99	NA	$280 > T_{avg} > 200$
5	Cold Shutdown [b]	< 0.99	NA	≤ 200
6	Refueling [c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE	The purpose of this section is to explain the meaning of logical connectors. Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.
BACKGROUND	Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors. When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.
EXAMPLES	The following examples illustrate the use of logical connectors.

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability. If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition. Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

(continued)

1.3 Completion Times

DESCRIPTION (continued)

However, when a subsequent train, subsystem, component, or variable, expressed in the Condition, is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

(continued)

1.3 Completion Times (continued)

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

The Completion Times of Conditions A and B are modified by a logical connector, with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (plus the extension) expires while one or more valves are still inoperable, Condition B is entered.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 4.	12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-5 (continued)

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR A.2 Place the channel in bypass.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-6 (continued)

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

(continued)

1.3 Completion Times

IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

EXAMPLES

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

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----NOTE----- Not required to be performed until 12 hours after ≥ 25% RTP. -----	
Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

CTS DISCUSSION OF CHANGES

ITS Section 1.0: Use and Application

ADMINISTRATIVE

A1 The designated change represents a non-technical, non-intent change to the Arkansas Nuclear One, Unit 1 Current Technical Specifications (CTS) made to make the ANO-1 Improved Technical Specifications (ITS) consistent with the Babcock and Wilcox (B&W) revised Standard Technical Specification (RSTS), NUREG 1430, Revision 1. This change does not alter the requirements of the CTS or RSTS. Examples of this type of change include: wording preference; convention adoption; editorial, numbering and formatting changes; and hierarchy structure.

A2 Not used.

A3 Not used.

A4 The RSTS establishes MODES of operation which are equivalent to the Reactor Operating Conditions defined in Section 1.2 of the CTS. The CTS presents individual definitions for each Reactor Operating Condition. The MODE equivalent of these Conditions will be defined by the combination of reactivity condition (Keff), % Rated Thermal Power, Average Reactor Coolant Temperature and bolting status of the reactor vessel head closure studs in the ITS (MODE definition and Table 1.1-1). The CTS defines the reactivity condition in terms of a subcritical condition (expressed in $\%\Delta k/k$). The RSTS defines the reactivity condition in terms of Keff. The ITS will adopt the Keff convention treating the small absolute difference between Shutdown Margin and Keff as a purely administrative change. In addition, the overlap of Cold Shutdown and Refueling is eliminated with the ITS definitions such that the unit is only in one of the defined MODES. The relocation of the CTS definitions for Reactor Operating Conditions into the ITS Table 1.1-1 is considered a purely administrative change. This change is consistent with the RSTS method of presentation of MODES. The applicability of the Reactor Operating Condition definition changes will be evaluated at each occurrence of the defined Reactor Operating Condition in the CTS. Changes to the CTS will be discussed on an individual basis with the Specification. Each change will be evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis.

A5 The CTS 1.2.1 reference to pressure in defining a Reactor Operating Condition is redundant to the requirements of CTS 3.1.2 which defines the allowable combination of Reactor Coolant System pressure and temperature. In establishing operational MODES in the ITS, the removal of the reference to pressure in defining a Reactor Operating Condition is considered an administrative change.

CTS DISCUSSION OF CHANGES

A6 CTS 1.2.2 defines Hot Shutdown in terms of a subcritical condition (1% $\Delta k/k$ shutdown) and an average reactor coolant temperature of greater than or equal to 525°F. This Hot Shutdown operating condition definition will be modified to correlate with the MODE 4 (Hot Shutdown) criteria established in RSTS Table 1.1-1. The RSTS MODE 4 criteria (per Table 1.1-1) imposes a maximum average reactor coolant temperature criteria of 280°F and a minimum average reactor coolant temperature of 200°F. The lower average reactor coolant temperature band could represent more restrictive requirements on the operation of the facility. Specifically, equipment that was previously required when average reactor coolant temperature exceeded 350°F may now be required when the average reactor coolant temperature exceeds 200°F. The applicability of this Reactor Operating Condition definition change will be evaluated at each occurrence of the defined Hot Shutdown Applicability in the CTS. Changes to the CTS will be discussed on an individual basis with the Specification. Each change will be evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis.

A7 CTS 1.2.4 which defines Hot Standby presently correlates to the RSTS MODE 2 (Startup) criteria. The CTS Hot Standby definition will be revised to correlate with the RSTS MODE 3 (Hot Standby) criteria. By adopting the RSTS convention, the CTS Hot Standby definition could impose more stringent requirements on the facility if this definition were substituted for the CTS Hot Standby in the Specification Applicability statements without consideration for the intent of the Specification (i.e. action to reduce reactor power level vice actions to take the reactor subcritical). For example, ACTIONS in the CTS that presently direct the unit to Hot Standby (which would allow critical operation at a power level below 2%) will now require that the reactor be taken to a subcritical condition ($K_{eff} < 0.99$). Similarly, during a plant heatup, the new MODE definition would require equipment to be placed into service at a lower operating temperature (280°F vice 350° or 525°F) than required by the CTS. The applicability of this Reactor Operating Condition definition change will be evaluated at each occurrence of the defined Hot Standby Applicability in the CTS. Changes to the CTS will be discussed on an individual basis with the Specification. Each change will be evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis.

A8 The CTS 1.3 definition of OPERABLE-OPERABILITY requires the capability of "*<u>necessary</u>* ... normal [(offsite)] <u>AND</u> emergency [(DG)] electrical power sources ... that are required for the system ... to perform its function(s)" (emphasis added). However, in MODES 1, 2, 3, and 4 CTS LCO 3.0.5 allows the features to be considered OPERABLE provided at least one source of power is still available and their redundant features are OPERABLE. In the ITS, the definition has been modified to require "normal <u>OR</u> emergency electrical power." For MODES 1, 2, 3, and 4, the CTS LCO 3.0.5 requirements are incorporated into the improved Technical Specification LCO 3.8.1 ACTIONS for when an emergency diesel generator or an offsite power source is inoperable. Thus, the ITS requirements are effectively the same as the current Technical Specification requirements.

For other than MODES 1, 2, 3, and 4 (i.e., "cold shutdown conditions"), the ANO-1 assumed and credited functions for safety related systems do not rely on offsite AND DG power. The "*necessary* ... power sources" are met with simply providing power from normal OR emergency sources. Therefore, the ITS presentation of the definition of OPERABLE-OPERABILITY fully captures the CTS definition for these shutdown conditions. Additionally, CTS 3.1.1.6 and 3.8.3.b and associated footnotes "**", explicitly reflected this assumption. This clarifying footnote is therefore deleted as it is consistent with the ITS definition. (Refer also to Section 3.8 of the ITS conversion submittal, Discussion of Difference #17 for related discussion on electrical power sources during shutdown conditions).

A9 The CTS 1.5.1 and 1.5.2 definitions for Trip Test and Channel Test, respectively, when combined, are considered to be equivalent to the RSTS definition of CHANNEL FUNCTIONAL TEST. Therefore, the CHANNEL FUNCTIONAL TEST definition from the RSTS has been adopted in its entirety. In addition, the sentence "The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is functionally tested" was added to provide clarification for how the test may be performed. The addition of this sentence represents the continuation of the current operating practice which would allow testing in this manner. Lastly, the addition of this sentence establishes consistency with the CHANNEL CALIBRATION definition given in the RSTS and adopted for use in the ITS.

A10 Selected definitions are deleted because the CTS that use these definitions are not retained in the ITS; or the equivalent ITS will not use the defined term. Discussions of the technical aspects of these changes are addressed in the discussion of change (DOC) for the individual specifications where the phrase is used in the CTS. The removal of a definition that is not used in the ITS is an administrative change because it has no impact on the implementation of any existing requirement not addressed in the ITS conversion. These deleted definitions are: CTS 1.2.3, 1.4, 1.5.5, 1.8, and 1.11 through 1.15.

A11 This administrative change adds definitions to the ITS that are established in the RSTS but which do not exist as definitions in the CTS. The addition of the definitions is made to make the ITS consistent with RSTS. The addition of the definitions by itself does not add limitations or requirements on the facility and is therefore considered to be an administrative change. These additional definitions are: MODES, ACTIONS, LEAKAGE, CONTROL RODS, AXIAL POWER SHAPING RODS, PHYSICS TESTS, THERMAL POWER, ALLOWABLE THERMAL POWER, and SHUTDOWN MARGIN.

A12 The CTS 1.2.5 definition for Power Operation makes specific reference to the power range channels (nuclear instruments) as representing the instrumentation used to determine the transition from CTS Reactor Operating Condition Hot Standby to Power Operation. The ITS will establish the transition from Startup (MODE 2) to Power Operation (MODE 1) as a function of percent RATED THERMAL POWER. This change in identification criteria is considered to be administrative because the nuclear instrumentation is calibrated to a heat balance which represents a measure of the thermal power of the reactor.

A13 CTS 1.2.5 establishes the transition power level between the Hot Standby and Power Operation Reactor Operating Conditions as 2% rated power as indicated on the power range channels (nuclear instrumentation). The ITS will establish the transition power level as 5% RATED THERMAL POWER in accordance with Table 1.1-1 of the RSTS. The 5% RTP MODE transition criteria is adopted for the purpose of maintaining consistency with the RSTS and with the ANO-2 Technical Specifications.

The different MODES are typically defined as transition points when more or less equipment is required to be operable. The accident analyses defined in the SAR are not impacted by this change in MODE transition. These accidents are based on worst case conditions and are not dependent on MODES, other than for the assumption of the equipment available to operate during an accident. NUREG-1430 has been reviewed for those instances in which additional equipment OPERABILITY is required as a result of entering MODE 1 from MODE 2 and MODE 2 from MODE 1. In the instance of the first MODE change, the following Specifications were found:

3.1.8 Physics Test Exceptions,

3.2.5 Power Peaking Factors, and

3.4.1 RCS Pressure, Temperature, and Flow DNB Limits.

In the instance of the second MODE change, the following Specifications were found:

3.3.9 Source Range Neutron Flux and

3.3.10 Intermediate Range Neutron Flux.

The CTS requirements for Physics Testing (3.1.8) are based on RCS pressure and not MODES. As stated in Item A4, these requirements will be evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis. LCO 3.2.5 is incorporated in the ANO ITS as 3.2.5, "Power Peaking" with an Applicability of MODE 1 with reactor power ≥ 20% RTP, as discussed in package section 3.2. Therefore, this difference in MODE 1 definition has no bearing with respect to LCO 3.2.5. The requirements of LCO 3.4.1 are not specified in the CTS and the inclusion of these requirements is considered to be more restrictive in total and a difference in MODE 1 definition has no bearing with respect to current requirements. The CTS requires OPERABILITY of the source and intermediate range neutron instrumentation during "startup and operation" while NUREG-1430 requires these instruments to be OPERABLE during MODES 2, 3, 4, and 5 (for source range) and MODE 2, When any CRD trip breaker is in the closed position and the CRD system is capable of rod withdrawal (for the intermediate range). The impact of the difference between 2% and 5% RTP on LCO 3.3.9 and 3.3.10 requirements will be evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis, as discussed in DOC A4.

A13 (continued)

NUREG-1430 was also reviewed for those instances in which a REQUIRED ACTION directs entry into MODE 2 from MODE 1. The following Specifications were found to apply:

3.2.5 Power Peaking Factors,

3.4.1 RCS Pressure Temperature, and Flow DNB Limits, and

3.7.2 MSIVs

As previously discussed for LCOs 3.2.5 and 3.4.1, the change from 2% to 5% RTP has no effect on the requirements. With respect to LCO 3.7.2, The CTS require the plant to be placed in Hot Shutdown in the event one MSIV is inoperable. For this same CONDITION, NUREG-1430 requires placing the plant in MODE 2. Again, the difference between 2% and 5% RTP has no bearing on the less restrictive nature of the MSIV requirements. The impact of the difference between 2% and 5% RTP on LCO 3.7.2 requirements will be evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis, as discussed in DOC A4.

Note: DOC A12 addresses the equivalence between the CTS reference to power range (nuclear) instrumentation and the ITS reference to RATED THERMAL POWER.

A14 The modification of CTS 1.2.6, Refueling Shutdown, to the RSTS equivalent MODE 6, Refueling, results in the deletion of the requirement that the reactor must be maintained subcritical by 1% dk/k even with all control rods removed and the coolant temperature at the decay heat removal pump suction is at the refueling temperature (normally 140°F). These conditions differ significantly from the RSTS Bases for LCO 3.9.1, Boron Concentration during Refueling Operations. The Bases for ITS LCO 3.9.1 state that the procedures establish a boron concentration that will maintain an overall core reactivity of Keff < 0.95 during fuel handling, with the control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures.

The RSTS definition for MODE 6, Refueling, in RSTS Table 1.1-1 will be adopted in the ITS. The review of RSTS 3.9.1 and its Bases will evaluate the implications of this change in definition and will categorize the adoption of RSTS 3.9.1 and its Bases as more restrictive or less restrictive as appropriate.

A15 CTS 1.9 currently defines Staggered Test Basis. The adoption of the RSTS definition for STAGGERED TEST BASIS in the ITS is considered an administrative change in that the required interval at which a component is actually surveilled is not changed. The manner of presentation in the Surveillance Requirements portion of the ITS will change; however, to reflect the RSTS definition. Further, each CTS which references a Staggered Test Basis will have to be individually evaluated and modified to reflect the formatting and presentation requirements of the RSTS definition.

A16 The CTS 1.5.6 definition for Heat Balance Calibration constitutes a specific application of a CHANNEL CALIBRATION to the power range nuclear instrumentation. In conformance with the terminology and format of the RSTS, the duplication of the term calibration will be eliminated through the consideration of the Heat Balance Calibration to be a type of CHANNEL CALIBRATION. This eliminates the need to retain the Heat Balance Calibration definition. [Note: The second portion of the CTS definition dealt with the methodology for the Heat Balance Calibration. As signified by the LATER indication, this information will be relocated into the Bases of ITS 3.3.1.]

A17 Not used.

A18 The CTS is revised to include ITS 1.2 which establishes the usage and convention for Logical Connectors used throughout the ITS. In addition, ITS 1.2 demonstrates through example the usage of the Logical Connectors. The ITS will adopt this usage and convention. This is an administrative change made to make the CTS conform to the NUREG-1430 convention.

A19 The CTS is revised to include ITS 1.3 which establishes the use and convention for Completion Times associated with the LCOs throughout the ITS. In addition, ITS 1.3 demonstrates through example the correct interpretation and usage of the Completion Times. The ITS will adopt this usage and convention. This is an administrative change made to make the CTS conform to the NUREG-1430 convention.

A20 The CTS is revised to include ITS 1.4 which establishes the use and convention for Frequency requirements associated with the Surveillance Requirements throughout the ITS. In addition, ITS 1.4 demonstrates through example the correct interpretation and usage of the Frequency requirements. The ITS will adopt this usage and convention. This is an administrative change made to make the CTS conform to the NUREG-1430 convention.

CTS DISCUSSION OF CHANGES

TECHNICAL CHANGE – MORE RESTRICTIVE

M None

TECHNICAL CHANGE – LESS RESTRICTIVE

L None

LESS RESTRICTIVE – ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 None

1.0 USE AND APPLICATION

1.1 Definitions

- - - - - - - - - - - - Note - - - - - - - - - - - -

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

- (A1)

~~I. Definitions~~

~~The following terms are defined for uniform interpretation of these specifications.~~ (A1)

RATED THERMAL POWER (RTP)

1.1 RATED THERMAL POWER (RTP)

Rated power is a steady state reactor core output of 2568 MWt. (A1)

1.2 ~~REACTOR OPERATING CONDITIONS~~ (A1)

Table 1.1-1 MODE 5 & Note (b)

1.2.1 Cold Shutdown <Apply Table 1.1-1; Note (b)> (A4)

The reactor is in the cold shutdown condition when it is subcritical by at least 1 percent $\Delta k/k$ and T_{avg} is no more than 200 F. ~~Pressure is defined by Specification 3.1.2.~~ (A5)

Table 1.1-1 MODE 4 & Note (b)

1.2.2 Hot Shutdown <Apply Table 1.1-1; Note (b)> (A4)

The reactor is in the hot shutdown condition when it is subcritical by at least 1 percent $\Delta k/k$ and T_{avg} is ~~at or~~ less than 280°F and greater than ~~525 F~~ 200°F. (A6)

~~1.2.3 Reactor Critical~~

~~The reactor is critical when the neutron chain reaction is self-sustaining and K_{eff} = 1.0.~~ (A10)

Table 1.1-1 MODE 3

1.2.4 Hot Standby (A4)

The reactor is in the hot standby condition when all of the following conditions exist:

A. T_{avg} is greater than ~~525 F~~ or equal to 280°F. (A7)

B. The ~~reactor is critical~~ reactivity condition is <0.99.

~~C. Indicated neutron power on the power range channels is less than 2 percent of rated power.~~

Table 1.1-1 MODE 1 & Note (a)

1.2.5 Power Operation <Apply Table 1.1-1; Note (a)> (A4)

The reactor is in a power operating condition when the indicated neutron power is above ~~2~~ 5 percent of rated power. ~~as indicated on the power range channels.~~ (A12) (A13)

Table 1.1-1 MODE 6 & Note (c)

1.2.6 Refueling ~~Shutdown~~ <Apply Table 1.1-1; Note (c)> (A4)

The reactor is in the refueling ~~shutdown~~ condition when one or more reactor vessel head closure bolts is/are less than fully tensioned. ~~even with all rods removed, the reactor would be subcritical by at least 1 percent $\Delta k/k$ and the coolant temperature at the decay heat removal pump suction is at the~~ (A14) refueling temperature (normally 140F). Pressure is defined by Specification 2.1.2. A refueling shutdown refers to a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods. A14 A5 A1

CORE ALTERATION

1.2.7 Refueling Operation <Add CORE ALTERATION> A1

An operation involving a change in core geometry by manipulation of fuel or control rods when the reactor vessel head is removed.

Table 1.1-1 MODE 2 & Note (a)

1.2.8 Startup <Apply Table 1.1-1, Note (a)> A4

The reactor shall be considered in the startup mode when the shutdown margin is reduced with the intent of going critical. reactivity condition is ≥ 0.99 and the THERMAL POWER is ≤ 5% RTP. A13

OPERABLE-OPERABILITY

1.3 OPERABLE - OPERABILITY

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s), and when ~~Implicit in this definition shall be the assumption that~~ all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its specified function(s) are also capable of performing their related support function(s). A1 or and A8 and A1

1.4 PROTECTION INSTRUMENTATION LOGIC A1

1.4.1 Instrument Channel A10

An instrument channel is the combination of sensor, wires, amplifiers and output devices which are connected for the purpose of measuring the value of a process variable for the purpose of observation, control and/or protection. An instrument channel may be either analog or digital.

1.4.2 Reactor Protection System A10

The reactor protection system is shown in Figures 7-1 and 7-9 of the FSAR. It is that combination of protective channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protection channels, their associated instrument channel inputs, manual trip switch, all rod drive control protective trip breakers and activating relays or coils.

A protection channel, as shown in Figure 7-1 of the FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply

<Add ACTIONS definition> A11

<Add MODE definition> A11

< Add LEAKAGE DEFINITION > A11

~~units, amplifiers and bistable modules provided for every reactor protection safety parameter, is a combination of instrument channels forming a single digital output to the protection system's coincidence logic. Each protection channel includes two key operated bypass switches, a protection channel bypass switch and a shutdown bypass switch.~~ A10

~~1.4.4 Reactor Protection System Logic~~

~~This system utilizes reactor trip module relays (coils and contacts) in all four of the protection channels as shown in Figure 7-1 of the FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one-out-of-two-times-two logic. Each element of the one-out-of-two-times-two logic is controlled by a separate set of two-out-of-four logic contacts from the four reactor protection channels.~~ A10

~~1.4.5 Safety Features System~~

~~This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in Figure 7-6 of the FSAR. The digital sub-system is wired to provide appropriate signals for the actuation of redundant safety features equipment on a two-of-three basis for any given parameter.~~ A10

~~1.4.6 Degree of Redundancy~~

~~The difference between the number of operable channels and the number of channels which, when tripped, will cause an automatic system trip.~~ A10

~~1.5 INSTRUMENTATION SURVEILLANCE~~ A1

1.5.1 Trip Test

A trip test is a test of logic elements in a protection channel to verify their associated trip action. A9

1.5.2 Channel FUNCTIONAL ~~Test~~ < CHANNEL FUNCTIONAL TEST DEFINITION AS PRESENTED IN THE ITS. > A9

CHANNEL FUNCTIONAL TEST

A channel test is the injection of an internal or external test signal into the channel to verify its proper response, including alarm and/or trip initiating action, where applicable.

1.5.3 ~~Instrument~~ Channel Check <CHANNEL CHECK DEFINITION>

CHANNEL CHECK

An instrument channel check is a verification of acceptable instrument performance by observation of its behavior and/or state; this verification includes comparison of output and/or state of independent channels measuring the same variable. A1

< Add CONTROL RODS DEFINITION > A11

< Add AXIAL POWER SHAPING RODS DEFINITION > A11

1.1

<Add PHYSICS TESTS definition> A11

<Add CHANNEL CALIBRATION definition>

1.5.4 ~~Instrument~~ Channel Calibration A1

~~An instrument channel calibration is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip and shall be deemed to include the channel test.~~

1.5.5 ~~Heat Balance Check~~ A10

~~A heat balance check is a comparison of the indicated neutron power and core thermal power.~~

1.5.6 ~~Heat Balance Calibration~~ A16

~~An adjustment of the power range channel amplifiers output to agree with the core thermal power as determined by a weighted primary and secondary heat balance considering all heat losses.~~ Between 0 and 15% power, only the primary heat balance is considered. From 15 to 100% power the heat balance is weighted linearly with only the secondary heat balance being considered at 100% power. LATER <LATER> (3.3A)

~~1.6 POWER DISTRIBUTION~~ A1

1.6.1 Quadrant Power Tilt CAPS (QPT) A1

QUADRANT POWER TILT (QPT)

~~Quadrant power tilt~~ QPT shall be defined by the following equation and is expressed as a percentage

$$QPT = 100\left(\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1\right)$$

1.6.2 ~~Reactor~~ AXIAL Power Imbalance CAPS A1

AXIAL POWER IMBALANCE

~~Reactor~~ AXIAL power imbalance CAPS ~~is~~ shall be the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of RATED THERMAL POWER (RTP) ~~rated power~~ RTP. ~~Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.~~

<Add THERMAL POWER definition> A11

<Add ALLOWABLE THERMAL POWER definition>

1.7 REACTOR BUILDING

<LATER> (3.6)

Reactor building integrity exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel lock and emergency lock are closed and sealed, or b. below.

b. At least one door on each of the personnel lock and emergency lock is closed and sealed during personnel access or repair.

c. All non-automatic reactor building isolation valves and blind flanges are closed as required.

d. All automatic reactor building isolation valves are operable or deactivated in the closed position.

e. The reactor building leakage determined at the last testing interval satisfies Specification 4.4.1.

LATER

1.8 FIRE SUPPRESSION WATER SYSTEM

~~The fire suppression water system consists of: water sources, pumps, and distribution piping with associated sectionalizing isolation valves. Such valves include the hose standpipe shutoff valves and the first valve ahead of the water flow alarm device on each sprinkler system.~~

A10

1.9 STAGGERED TEST BASIS

STAGGERED TEST BASIS

A staggered test basis shall consist of:

~~a. A test schedule for n systems, subsystems, trains or designated components obtained by dividing the specified test interval into n equal subintervals.~~

~~b. The testing of one system, subsystem, train or designated component at the beginning of each subinterval.~~

the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.

A15

1.1

1.10 **Dose Equivalent I-131**

Dose Equivalent I-131

The Dose Equivalent I-131 shall be the concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." — A1

1.11 **Liquid Radwaste Treatment System**

A Liquid Radwaste Treatment System is a system designed and used for holdup, filtration, and/or demineralization of radioactive liquid effluents prior to their release to the evironment.

1.12 **Purge - Purging**

Purge or Purging is the controlled process of discharging air or gas from a confinement to reduce the airborne radioactivity concentration in such a manner that replacement air or gas is required to purify the confinement.

1.13 **Member(s) of the Public**

Member(s) of the Public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

1.14 **Exclusion Area**

The exclusion area is that area surrounding ANO within a minimum radius of .65 miles of the reactor buildings and controlled to the extent necessary by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

1.15 **Unrestricted Area**

An unrestricted area shall be any area beyond the exclusion area boundary. — A10

1.16 **Core Operating Limits Report**

COLR

The CORE OPERATING LIMITS REPORT is the ANO-1 specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Technical Specification 6.12.3. Plant operation within these operating limits is addressed in individual specifications. — A1

5.6.5

<Add SHUTDOWN MARGIN Definition> — A11

(LATER) (3.4A)

1. Reactor Coolant Loop (A) and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and at least one associated reactor coolant pump.

Otherwise, restore the required loops to operable status within 72 hours or reduce the reactor coolant average temperature to less than or equal to 280°F within the next 12 hours.

B. With the reactor coolant average temperature above 280°F, at least one of the reactor coolant loops listed above shall be in operation.

Otherwise, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

3.1.1.6 Decay Heat Removal

With the reactor coolant average temperature at or below 280°F, but the reactor above the refueling shutdown condition, at least two of the coolant loops listed below shall be operable, and at least one loop shall be in operation:*

1. Reactor Coolant Loop (A) and its associated steam generator and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and its associated steam generator and at least one associated reactor coolant pump.

3. Decay Heat Removal Loop (A)**

4. Decay Heat Removal Loop (B)**

A8

A. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

B. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

*All reactor coolant pumps and decay heat removal pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**The normal or emergency power source may be inoperable when the reactor is in a cold shutdown condition.

A8

<Insert CTS 23A>

3.1.4 Reactor Coolant System Activity

Specification

<LATER> (3.4B)

3.1.4.1 Whenever the reactor is operating under steady-state conditions, the following conditions shall be met.

LATER

a. The total specific activity of the primary coolant shall not exceed 72/E μCi/gm where E is the sum of the average beta energy and average gamma energy per disintegration in MEV/disintegration.

Ē - AVERAGE DISINTEGRATION ENERGY

A1

b. The I-131 dose equivalent of the radioiodine activity in the primary coolant shall not exceed 3.5 μCi/gm.

c. If the radioactivity in the primary coolant exceeds the limits given above, corrective action shall be taken immediately to return the coolant activity to within these specifications. If the specific activity limits given above cannot be achieved within 24 hours, the reactor shall be brought to a hot shutdown condition using normal operating procedures. If the coolant radioactivity is not reduced to acceptable limits within an additional 48 hours, the reactor shall be brought to a cold shutdown condition and the cause of the out-of-specification operation ascertained.

Bases

<LATER> (3.4.B)

Rupture of a steam generator tube would allow primary coolant activity to enter the secondary coolant. The major portion of this activity is noble gases and would be released to the atmosphere from the condenser vacuum pump or a relief valve. Activity would continue to be released until the operator could reduce the primary system pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single steam generator tube, followed by isolation of the faulty steam generator within 34 minutes after the tube break. Assuming the full differential pressure across the steam generator, no more than one-quarter of the total primary coolant could be released to the secondary coolant in this period. The decay heat during this period of 1 hour for pressure reduction will generate steam in the secondary system representing less than 15 weight percent of the secondary system.

LATER

The parameters assumed in the dose analysis for the single steam generator tube failure included the following values:

1) total primary coolant volume (mass) = 5.2×10^5 lbs.

2) total secondary coolant volume (mass) = 2×10^6 lbs.

3) leakage rate from primary to secondary system = 1 gpm.

4) fission product decay heat energy for 1 hour = 1.56×10^8 BTU.

1.2
1.3
1.4

<INSERT CTS 23A>

< Add Section 1.2, Logical Connectors > A18

< Add Section 1.3, Completion Times > A19

< Add Section 1.4, Frequency > A20

1.1

3.8 FUEL LOADING AND REFUELING

LATER

Applicability

(LATER) (3.9)

Applies to fuel loading and refueling operations.

Objective

To assure that fuel loading, refueling and fuel handling operations are performed in a responsible manner.

Specification

3.8.1 Radiation levels in the reactor building refueling area shall be monitored by instrument RE-8017. Radiation levels in the spent fuel storage area shall be monitored by instrument RE-8009. If any of these instruments become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.

R TRM

3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, each with continuous indication available, whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

LATER

(LATER) (3.9)

3.8.3.a. At least one decay heat removal loop shall be in operation.* Otherwise, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the reactor coolant system, and close all containment penetrations providing access from the containment atmosphere to the outside atmosphere within 4 hours.

b. When the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet, two decay heat removal loops shall be operable**

A8

Otherwise, immediately initiate corrective action to return the required loops to operable status as soon as possible.

3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required for refueling shutdown.

3.8.5 Direct communications between the control room and the refueling personnel in the reactor building shall exist whenever changes in core geometry are taking place.

R TRM

*The decay heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of core alterations.

LATER

(LATER) (3.9)

**The normal or emergency power source may be inoperable for each shutdown cooling loop.

A8

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 1.0: Use and Application

Entergy Operations has evaluated these proposed Technical Specification changes and has determined that they involve no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as indicated below:

No unit specific "Less Restrictive" changes identified.

ITS DISCUSSION OF DIFFERENCES

ITS Section 1.0: Use and Application

1 DE I-131 - The DOSE EQUIVALENT I-131 markup reflects that ANO Unit-1 CTS 1.10 presently specifies that the dose conversion factors specified in TID-14844 be used in the determination of DOSE EQUIVALENT I-131. Therefore, the second reference provided in the RSTS is shown as deleted, or more appropriately, as not having been adopted. This change is consistent with current license basis.

2 Not used.

3 Not used.

4 PTLR - The definition of PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) is not adopted. ANO-1 will maintain the RCS Pressure and Temperature Curves and Limits in the ITS and will not implement a PTLR at this time. Since a PTLR is not implemented, the definition serves no purpose and has been deleted. This change is consistent with current license basis.

5 PHYSICS TESTS - The specific chapter reference in part "a." of the PHYSICS TESTS definition was deleted and the plant specific usage of SAR versus FSAR was incorporated. This change was made due to the non-standard nature of the ANO-1 SAR. Removal of the reference to a specific chapter simply insured that all physics testing referenced in the SAR were encompassed by this definition. This change is consistent with current license basis.

6 The definitions of EMERGENCY FEEDWATER INITIATION AND CONTROL (EFIC) RESPONSE TIME, ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME, and REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME were not incorporated. These terms and the referenced testing were not incorporated into ITS because they were not consistent with CTS . Response time testing of these systems, as required by specifications in NUREG-1430, is not required by CTS. This change is consistent with current license basis.

7 EFPD - Incorporates TSTF-125, Rev. 1.

8 CHANNEL FUNCTIONAL TEST - Incorporates TSTF-124.
CHANNEL CALIBRATION - Incorporates TSTF-124.

9 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR and NUCLEAR HEAT FLUX HOT CHANNEL FACTOR - will not be incorporated into the Definitions section of the ITS because these terms are not used in any specific ITS LCO. Consistent with current license basis and unit specific surveillance capability, ITS 3.2.5 will require that core linear heat rate (LHR) limits be maintained in accordance with the limits established in the COLR. This change is consistent with current license basis.

10 Not used.

ITS DISCUSSION OF DIFFERENCES

11 APSRs - The definition of AXIAL POWER SHAPING RODS (APSRs) has been modified to specify that these are the part-length control components. This specifically excludes the full length control components (regulating rods) when they are being used to control the axial power distribution of the reactor.

12 CHANNEL CALIBRATION - Incorporates TSTF-019.

13 LEAKAGE - The reference to injection in the definition of LEAKAGE was deleted for the purposes of clarification. LEAKAGE is generally associated with the escape of fluids from a system or boundary within which they are desired to be retained. The reference to "injection" within this context is unnecessarily confusing. This change is consistent with current license basis.

14 LEAKAGE - Incorporates TSTF-040 except as discussed in DOD 13.

15 Not used.

16 Not used.

17 La - As a result of a meeting between the NEI Tech Spec Task Force and the NRC Tech Spec Branch and Containment System Branch on October 18, 1995 concerning 10 CFR 50, Appendix J, Option B implementation, a definition of La is not adopted in the ITS. "La" will be described in the program description for the Reactor Building (Containment) Leak Rate Testing Program. This is consistent with current license basis.

18 Not used.

19 ITS 1.3 - The Example 1.3-6, Required Action A.2 was changed from "Reduce THERMAL POWER to ≤ 50% RTP" to "Place the channel in bypass." This change was made to provide a Required Action in A.2 which would not automatically be accomplished by performing the Required Action in B.1. This also provides a more representative and useful example that will be consistent with actions contained in Section 3.3.

20 Incorporates TSTF-205, Rev 3.

21 The definition of RATED THERMAL POWER is revised to retain the CTS usage of "steady state." This clarifies the definition and is consistent with the ANO-1 CTS and with NRC enforcement guidance concerning rated thermal power level control. The definition of ALLOWABLE THERMAL POWER is also revised for consistency.

CTS

1.0 USE AND APPLICATION

1.1 Definitions

---------------------------------NOTE---------------------------------

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition	
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.	N/A
ALLOWABLE THERMAL POWER	ALLOWABLE THERMAL POWER shall be the maximum steady state (21) reactor core heat transfer rate to the reactor coolant permitted by consideration of the number and configuration of reactor coolant pumps (RCPs) in operation.	N/A
AXIAL POWER IMBALANCE	AXIAL POWER IMBALANCE shall be the power in the top half of the core, expressed as a percentage of RATED THERMAL POWER (RTP), minus the power in the bottom half of the core, expressed as a percentage of RTP.	1.6.2
AXIAL POWER SHAPING RODS (APSRs)	APSRs shall be the part-length (11) control components used to control the axial power distribution of the reactor core. The APSRs are positioned manually by the operator and are not trippable.	N/A
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass ~~the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST.~~ all devices in the channel required for channel OPERABILITY and (20) Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. ~~Whenever a~~ (12)	1.5.4

(continued)

CTS

1.1 Definitions

CHANNEL CALIBRATION (continued)	~~sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.~~ (12) The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps ~~so that the entire channel is calibrated.~~ (20)	1.5.4
	~~The CHANNEL CALIBRATION shall also include testing of safety related Reactor Protection System (RPS), Engineered Safety Feature Actuation System (ESFAS), and Emergency Feedwater Initiation and Control (EFIC) bypass functions for each channel affected by the bypass operation.~~ (8)	
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.	1.5.3
CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY ~~including required alarms, interlocks, display, and trip functions.~~ of all devices in the channel required for channel OPERABILITY (20) ~~The ESFAS CHANNEL FUNCTIONAL TEST shall also include testing of ESFAS safety related bypass functions for each channel affected by bypass operation.~~ (8) <INSERT SENTENCE>	1.5.2
CONTROL RODS	CONTROL RODS shall be all full length safety and regulating rods that are used to shut down the reactor and control power level during maneuvering operations.	N/A
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE	1.2.7

(20) The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total steps.

(continued)

CTS

1.1 Definitions

CORE ALTERATION (continued)	ALTERATIONS shall not preclude completion of movement of a component to a safe position.	
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the ~~unit~~ ANO-1 specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific PARAMETER limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.	1.16
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~[~~Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites." ~~or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity"].~~ ①	1.10
Ē—AVERAGE DISINTEGRATION ENERGY	Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > [15] minutes, making up at least 95% of the total noniodine activity in the coolant.	3.1.4.1.a
~~EFFECTIVE FULL POWER DAY (EFPD)~~	~~EFPD shall be the ratio of the number of hours of production of a given THERMAL POWER to 24 hours, multiplied by the ratio of the given THERMAL POWER to the RTP. One EFPD is equivalent to the thermal energy produced by operating the reactor core at RTP for one full day.~~ ⑦	
~~EMERGENCY FEEDWATER INITIATION AND CONTROL (EFIC) RESPONSE TIME~~	~~The EFIC RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its EFIC actuation setpoint at the channel sensor until the emergency feedwater equipment is~~ ⑥	N/A

(continued)

CTS

1.1 Definitions

~~EMERGENCY FEEDWATER INITIATION AND CONTROL (EFIC) RESPONSE TIME (continued)~~

~~capable of performing its function (i.e., valves travel to their required positions, pumps discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.~~ 6

~~ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME~~

~~The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.~~ 6

~~L_a~~

~~The maximum allowable containment leakage rate, L_a, shall be [0.25]% of containment air weight per day at the calculated peak containment pressure (P_a).~~ 17

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except RCP seal water ~~injection or~~ leakoff), that is captured and conducted to collection systems or a sump or collecting tank; N/A 13

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

(continued)

CTS

1.1 Definitions

LEAKAGE (continued)

3. Reactor Coolant System (RCS) LEAKAGE ~~injection and~~ 13 through a steam generator (SG) to the Secondary System; 14

b. Unidentified LEAKAGE (except RCP seal water leakoff)

All LEAKAGE that is not identified LEAKAGE ~~or controlled LEAKAGE;~~ 14

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall. 1.2.1

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel. 1.2.2 1.2.4 1.2.5 1.2.6 1.2.8

~~NUCLEAR HEAT FLUX HOT CHANNEL FACTOR $F_Q(Z)$~~

~~$F_Q(Z)$ shall be the maximum local linear power density in the core divided by the core average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions.~~ 9

~~NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$)~~

~~($F^N_{\Delta H}$) shall be the ratio of the integral of linear power along the fuel rod on which minimum departure from nucleate boiling ratio occurs, to the average fuel rod power.~~

OPERABLE—OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s). 1.3

(continued)

CTS

1.1 Definitions (continued)

PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: the SAR; a. Described in ~~Chapter [14, Initial Test Program] of the FSAR,~~ b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.	N/A 5
~~PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~	~~The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."~~	4
QUADRANT POWER TILT (QPT)	QPT shall be defined by the following equation and is expressed as a percentage. $QPT = 100 \left(\frac{\text{Power in any Core Quadrant}}{\text{Average Power of all Quadrants}} - 1 \right)$	1.6.1 21
RATED THERMAL POWER (RTP)	RTP shall be a total Steady State reactor core heat transfer rate to the reactor coolant of ~~[2544]~~ 2568 MWt.	1.1
~~REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME~~	~~The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power is interrupted at the control rod drive trip breakers. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.~~	6

(continued)

CTS

1.1 Definitions (continued)

SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM; b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the (nominal zero power design level); and c. There is no change in APSR position.	N/A
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.	1.9
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.	N/A

Table 1.1-1 (page 1 of 1)
MODES

CTS

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)	
1	Power Operation	≥ 0.99	> 5	NA	1.2.5
2	Startup	≥ 0.99	≤ 5	NA	1.2.8
3	Hot Standby	< 0.99	NA	≥ ~~[330]~~ 280	1.2.4
4	Hot Shutdown[b]	< 0.99	NA	~~[330]~~ 280 > T_{avg} > ~~[200]~~ 200	1.2.2
5	Cold Shutdown[b]	< 0.99	NA	≤ ~~[200]~~ 200	1.2.1
6	Refueling[c]	NA	NA	NA	1.2.6

(a) Excluding decay heat. 1.2.5 & 1.2.8

(b) All reactor vessel head closure bolts fully tensioned. 1.2.1 & 1.2.2

(c) One or more reactor vessel head closure bolts less than fully tensioned. 1.2.6

CTS

1.0 USE AND APPLICATION

1.2 Logical Connectors

N/A

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors.

(continued)

CTS

N/A

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

CTS
N/A

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

CTS
N/A

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability. If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition. Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

(continued)

CTS
N/A

1.3 Completion Times

DESCRIPTION (continued)

However, when a subsequent train, subsystem, component, or variable, expressed in the Condition, is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

(continued)

CTS
N/A

1.3 Completion Times (continued)

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

(continued)

CTS
N/A

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

(continued)

CTS
NA

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

(continued)

CTS
N/A

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
C. One Function X train inoperable. AND One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. OR C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

(continued)

CTS
NA

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

The Completion Times of Conditions A and B are modified by a logical connector, with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

(continued)

CTS
N/A

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (plus the extension) expires while one or more valves are still inoperable, Condition B is entered.

(continued)

CTS
N/A

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

---NOTE---
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

(continued)

CTS
NA

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-5 (continued)

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x. OR A.2 ~~Reduce THERMAL POWER to ≤ 50% RTP.~~ Place the channel in bypass.	Once per 8 hours 8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

19

(continued)

CTS
NA

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-6 (continued)

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

(continued)

CTS
N/A

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop

(continued)

CTS
N/A

1.3 Completion Times

EXAMPLES	EXAMPLE 1.3-7 (continued) after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.
IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

CTS
NA

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

EXAMPLES

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

(continued)

CTS
N/A

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

CTS
N/A

1.4 Frequency

EXAMPLES (continued)

<u>EXAMPLE 1.4-2</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP <u>AND</u> 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "<u>AND</u>" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "<u>AND</u>"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

(continued)

CTS
N/A

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----NOTE----- Not required to be performed until 12 hours after ≥ 25% RTP. ----- Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

This Section Addresses the Following Specifications:

NUREG-1430	ANO-1 ITS	Title
2.1	2.1	Safety Limits
2.2	2.2	Safety Limit Violations

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be ≤ 5080 - (6.5×10^{-3} x (Burnup, MWD/MTU)°F) for TACO2 applications and ≤ 4642 - (5.8×10^{-3} x (Burnup, MWD/MTU)°F) for TACO 3 applications.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limits of 1.3 for the BAW-2 correlation and 1.18 for the BWC correlation.

2.1.1.3 In MODES 1 and 2, Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the Variable Low RCS Pressure-Temperature Protective Limits as specified in the COLR.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.1.3 is violated, restore RCS pressure and temperature within limits AND be in MODE 3 within 1 hour.

2.2.3 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance within limits AND be in MODE 3 within 1 hour.

2.2.4 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to ≤ 2750 psig within 5 minutes.

2.2.5 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that reactor core SLs ensure specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and abnormalities. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that the fuel centerline temperature stays below the melting temperature.

Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2 (Ref. 2) and BWC (Ref. 3) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady state operation, normal operational transients and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC).

The 95 percent confidence level that DNB will not occur is preserved by ensuring that the DNBR remains greater than the DNBR design limit based on the applicable CHF correlation for the core design. In the development of the applicable DNBR design limit, uncertainties in the core state variables, power peaking factors, manufacturing-related parameters, and the CHF correlation may be statistically combined to determine a statistical DNBR design limit. This statistical design limit protects the respective CHF design limit. Additional retained thermal margin may also be applied to the statistical DNBR design limit to yield a higher thermal design limit for use in establishing DNB-based core safety and operating limits. In all cases, application of statistical DNB design methods preserves a 95 percent probability at a 95 percent confidence level that DNB will not occur (Ref. 4).

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. The maximum fuel centerline temperatures are given by the relationships defined in SL 2.1.1.1 for the respective fuel designs and are dependent on whether the TACO2 (Ref. 5) or TACO3 (Ref. 6) analysis was utilized. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding-water (zirconium-water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding. The oxidized cladding then exists in a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and abnormalities. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience fuel centerline melting.

The RPS setpoints, in combination with all the LCOs, are designed to prevent any analyzed combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a DNBR of less than the DNBR limit and preclude the existence of flow instabilities (Ref. 7).

Automatic enforcement of these reactor core SLs is provided by the following:

a. RCS High Pressure trip;

b. RCS Low Pressure trip;

c. Nuclear Overpower trip;

d. RCS Variable Low Pressure trip (also known as Pressure Temperature Trip);

e. Reactor Coolant Pump to Power trip;

f. Nuclear Overpower RCS Flow and AXIAL POWER IMBALANCE trip; and

g. RCS High Temperature trip.

The SL represents a design requirement for establishing the RPS trip setpoints identified previously.

SAFETY LIMITS

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature stays below the melting point, or the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or the exit quality is within the limits defined by the DNBR correlation. In addition, the COLR identifies the pressure/temperature operating region that keeps the reactor from reaching an SL when operating up to design power.

The COLR presents the most limiting condition of pressure/temperature combinations for all possible reactor coolant pump maximum THERMAL POWER combinations. Analyses have been performed which bound the three pump and two pump (one pump in each loop) allowed operating conditions based on the expected minimum flow rates and maximum ALLOWABLE THERMAL POWER for these operating conditions.

The SLs are preserved by monitoring the process variable AXIAL POWER IMBALANCE to ensure that the core operates within the fuel design criteria. AXIAL POWER IMBALANCE protective limits are preserved by their corresponding RPS setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," and are provided in the COLR. The trip setpoints are derived by adjusting the measurement system independent AXIAL POWER IMBALANCE protective limits given in the COLR to allow for measurement system observability and instrumentation errors.

The AXIAL POWER IMBALANCE protective limits are separate and distinct from the AXIAL POWER IMBALANCE operating limits defined by LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits." The AXIAL POWER IMBALANCE operating limits in LCO 3.2.3, also specified in the COLR, preserve initial conditions of the safety analyses but are not reactor core SLs.

APPLICABILITY

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. Automatic protection actions serve to prevent RCS heatup to reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.

In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1 AND 2.2.2

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the requirement to go to MODE 3 places the plant in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the plant to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

2.2.5

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 8).

REFERENCES

1. SAR, Section 1.4, GDC 10.
2. BAW-10000A, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," Babcock & Wilcox, Lynchburg, VA, May 1976 .
3. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," Babcock & Wilcox, Lynchburg, VA, April 1985.
4. BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Rev. 2, Babcock & Wilcox, Lynchburg, VA, October 1997.
5. BAW-10141P-A, Rev. 1, "TACO2 Fuel Pin Performance Analysis," Babcock & Wilcox, Lynchburg, VA, June 1983.
6. BAW-10162P-A, "TACO3 Fuel Pin Thermal Analysis Code," Babcock & Wilcox, Lynchburg, VA, October 1989.
7. SAR, Chapters 3 & 14. .
8. 10 CFR 50.72.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

In SAR, Section 1.4 (Ref. 1), GDC 14, "Reactor Coolant Pressure Boundary (RCPB)," and GDC 15, "Reactor Coolant System Design", address RCPB design and protection, respectively. The ANO-1 discussion regarding how GDC 15 is accomplished states that analysis and evaluation of all normal and abnormal operating conditions and transients are integrally related to all RCS and associated systems design. SAR Chapter 14 (Ref. 2) lists these abnormal operating conditions and transients and terms them "abnormalities". In addition, GDC 28, "Reactivity Limits" (Ref. 1), specifies that reactivity accidents including rod ejection do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psig. During normal operation and abnormalities, the RCS pressure is kept from exceeding the design pressure by more than 10% in order to remain in accordance with the design codes (Ref. 3 and 4). Hence, the safety limit is 2750 psig. To ensure system integrity, all RCS components were hydrostatically tested at 125% of design pressure prior to initial operation, according to the design code requirements. Inservice leak testing at not less than 2155 psig is also required, prior to MODE 2, following any opening of the reactor coolant system in accordance with ASME code, Section XI; IWA-5000. When performed at the end of refueling outages, this leak test also satisfies the requirements of IWB-2500, Table IWB-2500-1; Category B-P items B15.10, B15.20, B15.30, B15.40, B15.50, B15.60, and B15.70 for all Class I pressure retaining components (Ref. 5).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, operating in conjunction with the Reactor Protection System trip settings, ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME code for Nuclear Power Plant Components (Ref. 3). The design basis transient that is most influential for establishing the required relief capacity, and hence the valve size requirements and lift settings, is a rod withdrawal event from low power.

The startup event (rod withdrawal at low power) (Ref. 2) is performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

a. Electromatic relief valve (ERV);

b. Steam line turbine bypass valves;

c. Control system runback of reactor and turbine power; and

d. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS B31.7 (Ref. 4), is 110% of design pressure. Therefore, the SL on maximum allowable RCS pressure is 2750 psig.

Overpressurization of the RCS can result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 6).

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES during overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized significantly.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the RCS pressure SL.

2.2.3

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 6).

The allowed Completion Time of 1 hour is based on the importance of reducing power level to a MODE where the potential for challenges to safety systems is minimized.

2.2.4

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the RCS pressure SL in MODE 3, 4, or 5 is potentially more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

2.2.5

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 7).

REFERENCES

1. SAR, Section 1.4, GDC 14, GDC 15, and GDC 28, 1988.
2. SAR, Chapter 14.
3. ASME Boiler and Pressure Vessel Code, Section III, 1965-S67, Article NB-7000.
4. USAS B31.7, Nuclear Power Piping, 1969.
5. ASME Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
6. 10 CFR 100.
7. 10 CFR 50.72.

CTS DISCUSSION OF CHANGES

ITS Section 2.0: Safety Limits

ADMINISTRATIVE

A1 The designated change represents a non-technical, non-intent change to the Arkansas Nuclear One, Unit 1 Current Technical Specifications (CTS) made to make the ANO-1 Improved Technical Specifications (ITS) consistent with the Babcock and Wilcox (B&W) revised Standard Technical Specification (RSTS), NUREG 1430, Revision 1. This change does not alter the requirements of the CTS or RSTS. Examples of this type of change include: wording preference; convention adoption; editorial, numbering and formatting changes; and hierarchy structure.

A2 The CTS Bases will be administratively deleted in their entirety in favor of the NUREG-1430 Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 The requirement of CTS 6.7.1.b. to submit a report to the NRC "pursuant to the requirements of 10 CFR 50.36" was removed. This requirement is a duplication of the requirement found in 10 CFR.50.36 "Technical Specifications" paragraph (c)(1) and as such was redundant. The removal of this requirement from the CTS was administrative in nature because this requirement was contained elsewhere, namely 10 CFR 50.36.

A4 This page is not yet approved in its current form. Therefore, this markup is dependent on the expected NRC approval of the July 14, 1999, license amendment request (Ref. 0CAN079901) related to the post accident sample system.

TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 CTS 2.1.1, 2.1.2, & 2.1.3 establish the APPLICABILITY for the Reactor Core Safety Limits as "when the reactor is critical." ITS 2.1.1 will establish APPLICABILITY as MODES 1 and 2 which include Keff greater than or equal to 0.99. Thus, MODE 2 is more restrictive than CTS since it does not become applicable until Keff = 1.0. The additional Applicability is included because limiting accidents and transients are postulated which begin in this MODE. This requirement is consistent with NUREG-1430.

M2 CTS 2.2 does not establish required actions should the RCS Pressure Safety Limit be violated in MODES 3, 4, and 5. Therefore, the required actions of RSTS 2.2.4 are adopted in the ITS. The information shown as inserted on the CTS mark-up as ITS 2.2.4 represents more restrictive requirements than those presently imposed.

M3 CTS 6.7.1.a required that the Unit be placed in hot shutdown within one hour following the violation of a CTS defined Safety Limit. ITS 2.2.1, 2.2.2, and 2.2.3 will require that the Unit be placed in MODE 3. The ITS requirement is more restrictive in that it will require that the Unit have a Keff value of less than 0.99. The CTS requires that the Unit be taken 1% Δk/k subcritical. The Keff requirement is 0.01% Δk/k more restrictive.

TECHNICAL CHANGE – LESS RESTRICTIVE

L1 CTS 2.2.1 establishes APPLICABILITY for the RCS Pressure Safety Limit as being "when there are fuel assemblies in the reactor vessel." ITS 2.1.2 will establish APPLICABILITY as MODES 1, 2, 3, 4 & 5. In essence, the ITS would be marginally less restrictive as it would not apply during MODE 6 while the CTS would apply after the first assembly was placed in the vessel. Although a short time period may exist between MODE 5 and reactor vessel head removal in MODE 6, during which the Safety Limit will no longer apply, the consequences of a postulated overpressure event are mitigated by the implementation of low temperature overpressurization protection requirements and administrative controls.

LESS RESTRICTIVE – ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to the Bases. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Safety Limit, but rather describe an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications. The specific relocations are:

CTS Location	New Location
2nd sentence of SL 2.1.1	B 2.1.1 Applicable Safety Analyses
2nd sentence of SL 2.1.2	B 2.1.1 Applicable Safety Analyses
4.3.2	B 2.1.2 Background

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

A1

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

2.1 APPL Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow when the reactor is critical. M1

MODES 1 and 2

Objective

To maintain the integrity of the fuel cladding. A1

Specification

2.1.1.1 2.1.1 The maximum local fuel pin centerline temperature shall be $\leq 5080 - (6.5 \times 10^{-3} \times (\text{Burnup, MWD/MTU})\,°F)$ for TACO2 applications and $\leq 4642 - (5.8 \times 10^{-3} \times (\text{Burnup, MWD/MTU})\,°F)$ for TACO3 applications. edit Operation within this limit is ensured by compliance with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR. LA1 Bases

2.1.1.2 2.1.2 The departure from nucleate boiling ratio shall be maintained greater than the limits of 1.3 for the BAW-2 correlation and 1.18 for the BWC correlation. Operation within this limit is ensured by compliance with Specification 2.1.3 and with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR. LA1 Bases

2.1.1.3 2.1.3 Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the Variable Low RCS Pressure-Temperature Protective Limits as specified in the COLR.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which could result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2(1) and BWC(2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC). A2

A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure for the allowable RC pump combination has been considered in determining the Variable Low RCS Pressure-Temperature Protective Limits.

A2

The Variable Low RCS Pressure-Temperature Protective Limits presented in the COLR represent the conditions at which the DNBR is greater than or equal to the minimum allowable DNBR for the limiting combination of thermal power and number of operating reactor coolant pumps which is based on the nuclear power peaking factors (3) as specified in the COLR with potential fuel densification effects.

The Axial Power Imbalance Protective Limits in the COLR are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The DNBR limit produced by the limiting combination of the radial peak, axial peak, and position of the axial peak.

2. The combination of radial and axial peak that prevents central fuel melting at the hot spot as given in the COLR.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The flow rates for the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop.

The Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive of all possible reactor coolant pump maximum thermal power combinations as specified in the COLR. The Variable Low RCS Pressure-Temperature Protective Limits in the COLR represent the conditions at which the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation. If the actual pressure/temperature point is below and to the right of the pressure/temperature line, the Variable Low RCS Pressure-Temperature Protective Limit is exceeded. The local quality at the point of minimum DNBR is less than 22 percent (BAW-2)(1) or 26 percent (BWC)(2).

2.0

Using a local quality limit of 22 percent (BAW-2) or 26 percent (BWC) at the point of minimum DNBR as a basis for less than four reactor coolant pumps operating of the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the BAW-2 or the BWC correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

The maximum thermal power, as a function of reactor coolant pump operation is limited by the power level trip produced by the flux-flow ratio (percent flow x flux-flow ratio), plus the appropriate calibration and instrumentation errors.

For each combination of operating reactor coolant pumps of the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (BAW-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (BAW-2) or 26 percent (BWC) for that particular reactor coolant pump combination. The Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive because any pressure-temperature point above and to the left of this curve will be above and to the left of the other curves.

A2

REFERENCES

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, May, 1976.

(2) BWC Correlation of Critical Heat Flux, BAW-10143P-A, April, 1985.

(3) FSAR, Section 3.2.3.1.1.c.

2.0

(SLs)

2.2 SAFETY LIMITS - ~~REACTOR SYSTEM PRESSURE~~ A1

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity. A1

Specification

2.1.2 2.2.1 The reactor coolant system pressure shall not exceed 2750 psig ~~when there are fuel assemblies in the reactor vessel.~~ {MODES 1, 2, 3, 4, and 5}
2.1.2 Appl L1

<LATER> (3.4B) 2.2.2 The setpoint of the pressurizer code safety valves shall be in accordance with ASME, Boiler and Pressurizer Vessel Code, Section III, Article 9, Summer 1968. LATER

Bases

The reactor coolant system (1) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME code, Section III, is 110 percent of design pressure.(2) The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 is 110 percent of design pressure. Thus, the safety limit of 2750 psig (110 percent of the 2500 psig design pressure) has been established. (2) The settings for the reactor high pressure trip (2355 psig) and the pressurizer code safety valves (2500 psig ±1%)(3) have been established to assure that the reactor coolant system pressure safety limit is not exceeded. When testing the pressurizer code safety valves, the "as found" lift setpoint may be 2500 psig +1, -3%. However, if found outside of a ±1% tolerance band, they shall be reset to 2500 psig ±1%. The initial hydrostatic test is conducted at 3125 psig (125 percent of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2450 psig.(4) A2

REFERENCES

(1) FSAR, Section 4

(2) FSAR, Section 4.3.11.1

(3) FSAR, Section 4.2.4

(4) FSAR, Table 4-1

⟨LATER⟩ (5.0)

4.3 TESTING FOLLOWING OPENING OF SYSTEM

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To assure Reactor Coolant System integrity prior to return to criticality following normal opening, modification, or repair.

Specification

4.3.1 When Reactor Coolant System repairs or modifications have been made, these repairs or modifications shall be inspected and tested to meet all applicable code requirements prior to the reactor being made critical.

LATER

4.3.2 Following any opening of the Reactor Coolant System, it shall be leak tested at not less than 2155 psig, prior to the reactor being made critical, in accordance with the ASME Boiler and Pressure Vessel Codes, Section XI; IWA-5000.

LA1 Bases

⟨LATER⟩ (5.0)

4.3.3 The limitations of Specification 3.1.2 shall apply.

LATER

Bases

Repairs or modifications made to the Reactor Coolant System are inspectable and testable under applicable codes, such as B 31.7, and ASME Boiler and Pressure Vessel Code, Section XI.

For normal opening, the integrity of the Reactor Coolant System in terms of strength, is unchanged. The ASME Boiler and Pressure Vessel Code, Section XI; IWA-5000 requires a system leak test at nominal operating pressure (2155 psig) following system opening. At the end of refueling outages, this test also satisfies the requirements of IWB-2500, Table IWB-2500-1; Category B-P items B15.10, B15.20, B15.30, B15.40, B15.50, B15.60, and B15.70 for all Class I pressure retaining components.

A2

REFERENCES

(1) FSAR, Section 4
(2) ASME Boiler and Pressure Vessel Code, Section XI

6.6 DELETED

2.2 6.7 ~~SAFETY LIMIT VIOLATION~~ SL Violations A1

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

2.2.1, 2.2.2, 2.2.3 a. The facility shall be placed in at least ~~hot shutdown~~ MODE 3 within one hour. M3

2.2.5 b. within 1 hour The Nuclear Regulatory Commission shall be notified pursuant to 10 CFR 50.72 ~~and a report submitted pursuant to the requirements of 10 CFR 50.36 and Specification 6.6.~~ A1 A3

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. (Deleted)

e. (Deleted)

f. Fire Protection Program Implementation.

g. New and spent fuel storage.

h. Offsite Dose Calculation Manual and Process Control Program implementation at the site.

<LATER> (5.0) LATER

2.2.4 In MODES 3, 4 and 5, if SL 2.1.2 is violated, restore RCS pressure to ≤ 2750 psig within 5 minutes M2

LAR A4

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 2.0: Safety Limits

Entergy Operations has evaluated these proposed Technical Specification changes and has determined that they involve no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as indicated below:

<u>2.0 L1</u>

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

The change results in a modification of the Applicability of the Safety Limits. The Safety Limits are not accident initiators. Therefore, the probability of any previously evaluated accident is not significantly increased. The accident mitigation features of the plant are not affected by this change. Following implementation of this change, the reactor coolant system (RCS) Safety Limit must be met in MODES 1, 2, 3, 4, and 5. The current Applicability is stated as "when there are fuel assemblies in the vessel." This change results in a relaxation of the Applicability in that during MODE 6 the Safety Limit will no longer apply. Although a short time period may exist between entry into MODE 6 (when the first reactor vessel head bolt is detensioned), and actual reactor vessel head removal (following which overpressurization is not possible), the consequences of an overpressure event are mitigated by the implementation of low temperature overpressurization protection requirements and administrative controls. Therefore, the consequences of any previously evaluated accident are not significantly increased.

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

The Safety Limits are not accident initiators. Therefore, the scope of the change does not establish a potential new accident precursor.

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

This change does involve an incremental reduction in the margin of safety since the RCS pressure Safety Limit will no longer be applicable when fuel is in the reactor vessel and the unit is in MODE 6. However, this reduction is not considered significant in that sufficient controls exist to prevent the occurrence of and mitigate the effects of postulated low temperature overpressure events.

ITS DISCUSSION OF DIFFERENCES
ITS Section 2.0: Safety Limits

1 NUREG 2.1.1.1 - The plant specific information from CTS 2.1 for maximum local fuel pin centerline temperature was inserted in ITS 2.1.1.1. Two separate temperatures were inserted to account for the two analyzed fuel assembly types used at ANO-1. This information is consistent with the current licensing basis.

2 NUREG 2.1.1- Incorporates TSTF-126.

3 NUREG 2.2- Incorporates TSTF-005, Rev 1 with the exception that NUREG 2.2.5 is retained as a unit specific preference. This requirement is consistent with the current licensing basis.

4 NUREG 2.2 - The wording in ITS 2.2.3 and 2.2.4 was modified to be consistent with the wording used in ITS 2.2.1 and 2.2.2. The words "not met" were replaced with the word "violated." This change precludes the potential misinterpretation of an unintended distinction, is administrative in nature and has been made for consistency with similar ITS.

5 Bases - Reference to the Main Steam Safety Valves (MSSVs) as contributors in preventing the violation of Reactor Core Safety Limits was deleted at each occurrence. Chapter 3 and 14 of the Unit 1 SAR do not explicitly credit the MSSVs as functioning to prevent exceeding Reactor Core Safety Limits, since the startup evaluation does not model the secondary side.

Reference to the RCS High Temperature trip as a contributor in preventing the violation of Reactor Core Safety Limits was added. Although Chapter 3 and 14 of the Unit 1 SAR do not explicitly credit this trip function, it is relied upon to set boundaries for the analyses.

6 Bases - ANO-1 uses the terms "RCS Variable Low Pressure trip" and "Pressure Temperature trip" interchangeably. Therefore, both terms are presented in the Bases.

7 Bases - Specific detail relating to the two critical heat flux correlations at ANO-1 has been included in the ITS B 2.1.1 Background information. This information is consistent with the ANO-1 current licensing basis. References 2 and 3 have been added to reference the respective topical reports associated with the heat flux correlations.

8 Bases - Specific reference to the ASME code was deleted in favor of reference to "design codes" which more accurately reflects the number of codes to which the plant was designed and built.

9 Bases - The word "event" was added in paragraph two (2) of the APPLICABLE SAFETY ANALYSES section to more clearly define the basis for the relief valve capacity. This wording is consistent with the wording in the SAR which provides evaluation of individual rod, multiple rod and rod bank events. The last sentence on

page B2.0-6 of the BWOG STS was deleted as it does not accurately establish the plant conditions established in the ANO-1 SAR Safety Analyses supporting the determination of required relief valve capacity. These plant conditions are established in the ANO-1 SAR.

10 Bases - The ANO-1 Design Code for piping, valves and fittings was USAS B31.7 which provides for a maximum transient pressure of 110% of design pressure. Because this is the same allowance as stated under the ASME Code, Section III, the sentence starting with "The most limiting of these..." is unnecessary as both are equally limiting. In addition, the text cites Reference 6 which was also modified to accurately reflect the correct design code.

11 Bases - Power operated relief valve (PORV) has been replaced by the ANO-1 specific designation "electromatic relief valve (ERV). This change was made for consistency with ANO-1 documentation.

12 Bases - The background discussion for LCO 2.1.2 has been revised to incorporate the ANO-1 current licensing basis with respect to reactor coolant system (RCS) leak testing. Specifically, the NUREG-1430 description of the RCS inservice operational hydrotest at 100% design pressure has been replaced with a description of the CTS 3.3.2 RCS leak test performed at not less than 2155 psig. Information from the CTS 4.3.2 Bases describing other requirements satisfied by the performance of this leak test has also been included.

13 Bases - Specific detail was added to item 2 of the RCS Pressure Safety Limit REFERENCES specifying that the 1965, Summer '67 Addenda was the reference ASME Boiler and Pressure Vessel Code , Section III, used for determining the design requirements for the RCS pressurizer safety valves for ANO-1.

14 Bases - The Insert adds specific reference to the analysis code (TACO2 or TACO3) used in the fuel design analysis for determining the maximum fuel centerline temperature. This analysis is performed in accordance with the calculational methods described in BAW-10141 or BAW 10162 which were cited as references in section B 2.1.1.

15 Bases - The term AOO is used in the GDCs, but the ANO-1 licensing basis is contingent upon discussion of "abnormalities" as defined and listed in SAR, Section 14.1. The ANO-1 SAR was written partially based on the guidance given in a "Guide to the Organization and Contents of Safety Analysis Reports" issued by the Atomic Energy Commission on June 30, 1966. This document discusses what transients or "abnormalities" should be considered for Core and Coolant Boundary Protection Analysis. Statements concerning the GDC criteria are modified in the ITS to reference the current licensing basis description in the Unit 1 SAR.

ITS DISCUSSION OF DIFFERENCES

16 Bases - The word "significantly" is added to the last sentence of the Applicability discussion for 2.1.2. This is added to clarify that some pressurization due to the formation steam can be expected if the head is in place and not fully detensioned and removed. However, in agreement with the RSTS bases, the amount of pressurization is not expected to be significant and thus the Specification should not be applicable in MODE 6.

17 NUREG 2.2 - ITS 2.2.2 and 2.2.3 were editorially changed to reflect a Logical Connector structure consistent with the requirements of Section 1.2 of the ITS.

18 Bases - For ANO-1, the startup event (rod withdrawal from low power) is the limiting event for Pressurizer Safety Valve design; and thus, the Bases were modified to identify that this was the limiting event. The cited overpressure protection analyses were not the bases used and reference to them was deleted.

19 NUREG 2.1.1.3 is revised to retain the reference to the "Variable Low RCS Pressure/Temperature Protection Limits as specified in the COLR." This limit is maintained in the COLR (as recently approved in Amendment No. 186) since it is a cycle specific parameter. Use of this reference to the COLR also eliminates the need for NUREG Figure 2.1.1-1. The Bases are also revised to reference the COLR rather than the safety limit of ITS 2.1.1.3 since the COLR actually provides the pressure/temperature relationship. This modification improves clarity by providing a direct reference to the location of the limits. Additionally, a Bases paragraph is incorporated to establish that the COLR represents the most limiting condition of pressure/temperature combinations for reactor coolant pump maximum THERMAL POWER combinations. Analyses have been performed for three pump operations and one pump per loop operations which demonstrate the four pump curve is bounding. Incorporation of this statement clarifies the acceptability of operation with less than four RCPs.

20 Bases – Information related to Statistical Core Design (SCD) methodology has been added to maintain consistency with other Unit 1 LBDs. SCD was first integrated into the reload process for protection from DNB for the first time in Cycle 15. This method is described fully in topical report BAW-10187P-A and referenced in the reload methodology topical BAW-10179P-A. A reference to BAW-10179P-A has also been incorporated into the References Section.

CTS

2.0 SAFETY LIMITS (SLs)

5080 − (6.5 x 10⁻³ x (BURNUP, MWD/MTU) °F) for TACO2 applications AND ≤ 4642 − (5.8 x 10⁻³ x (BURNUP MWD/MTU) °F) for TACO3 applications. (1)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be ≤ ~~[5080 − (6.5 x 10⁻³ MWD/MTU) °F].~~ ~~Operation within this limit is ensured by compliance with the AXIAL POWER IMBALANCE protective limits preserved by the Reactor Protection System setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," as specified in the COLR.~~ (2) 2.1.1

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limits of ~~[~~1.3 for the BAW-2 correlation and 1.18 for the BWC correlation~~]~~. ~~Operation within this limit is ensured by compliance with SL 2.1.1.3 and with the AXIAL POWER IMBALANCE protective limits preserved by the RPS setpoints in LCO 3.3.1, as specified in the COLR.~~ (2) 2.1.2

2.1.1.3 In MODES 1 and 2, Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the ~~SL shown in Figure 2.1.1-1.~~ Variable Low RCS Pressure-Temperature Protective Limits as specified in the COLR. (19) 2.1.3

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ ~~[2750]~~ 2750 psig. 2.2.1

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour. 6.7.1.a

2.2.2 In MODE 1 or 2, if SL 2.1.1.3 is violated, restore RCS pressure and temperature within limits ~~and~~ AND be in MODE 3 within 1 hour. 6.7.1.a
(17)

(continued)

CTS

2.0 SLs

2.2 SL Violations (continued)

2.2.3 In MODE 1 or 2, if SL 2.1.2 is ~~not met~~ violated, restore compliance within limits ~~and~~ AND be in MODE 3 within 1 hour. (4) (1) (4) 6.7.1.a

2.2.4 In MODES 3, 4, and 5, if SL 2.1.2 is ~~not met~~ violated, restore RCS pressure to ≤ ~~[2750]~~ 2750 psig within 5 minutes. N/A

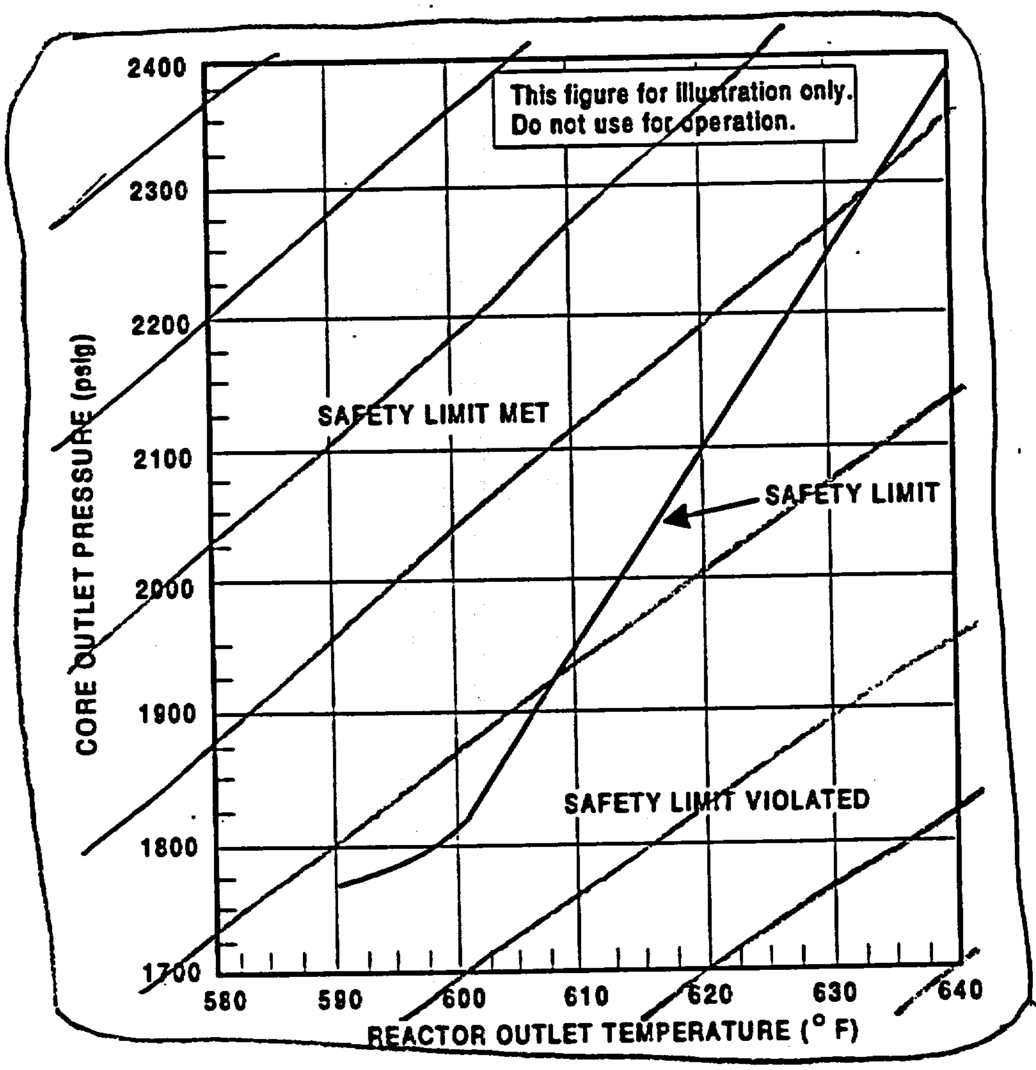
2.2.5 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72. 6.7.1.b

~~2.2.6 Within 24 hours, notify the [Vice President—Nuclear Operations].~~

~~2.2.7 Within 30 days, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC and the [Plant Superintendent, and Vice President—Nuclear Operations].~~

~~2.2.8 Operation of the plant shall not be resumed until authorized by the NRC.~~

(3)



19

~~Figure 2.1.1-1 (page 1 of 1)~~
~~Reactor Coolant System Departure from Nucleate Boiling Safety Limits~~

CTS

2.0 SAFETY LIMITS (SLs)

5080 - (6.5 x 10^{-3} x (BURNUP, MWD/MTU) °F) for TACO2 applications AND ≤ 4642 - (5.8 x 10^{-3} x (BURNUP, MWD/MTU) °F) for TACO3 applications. (1)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be ≤ ~~[5080 - (6.5 x 10^{-3} MWD/MTU) °F].~~ ~~Operation within this limit is ensured by compliance with the AXIAL POWER IMBALANCE protective limits preserved by the Reactor Protection System setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," as specified in the COLR.~~ 2.1.1 (2)

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limits of ~~[~~1.3 for the BAW-2 correlation and 1.18 for the BWC correlation~~]~~. ~~Operation within this limit is ensured by compliance with SL 2.1.1.3 and with the AXIAL POWER IMBALANCE protective limits preserved by the RPS setpoints in LCO 3.3.1, as specified in the COLR.~~ 2.1.2 (2)

2.1.1.3 In MODES 1 and 2, Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the ~~SL shown in Figure 2.1.1-1.~~ 2.1.3

Variable Low RCS Pressure-Temperature Protective Limits as specified in the COLR. (19)

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ ~~[2750]~~ 2750 psig. 2.2.1

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour. 6.7.1.a

2.2.2 In MODE 1 or 2, if SL 2.1.1.3 is violated, restore RCS pressure and temperature within limits ~~and~~ AND be in MODE 3 within 1 hour. 6.7.1.a (17)

(continued)

CTS

2.0 SLs

2.2 SL Violations (continued)

violated (4)

2.2.3 In MODE 1 or 2, if SL 2.1.2 is ~~not met~~ violated, restore compliance within limits ~~and~~ AND be in MODE 3 within 1 hour. (1) (4) 6.7.1.a

violated (4)

2.2.4 In MODES 3, 4, and 5, if SL 2.1.2 is ~~not met~~ violated, restore RCS pressure to ≤ ~~[2750]~~ 2750 psig within 5 minutes. N/A

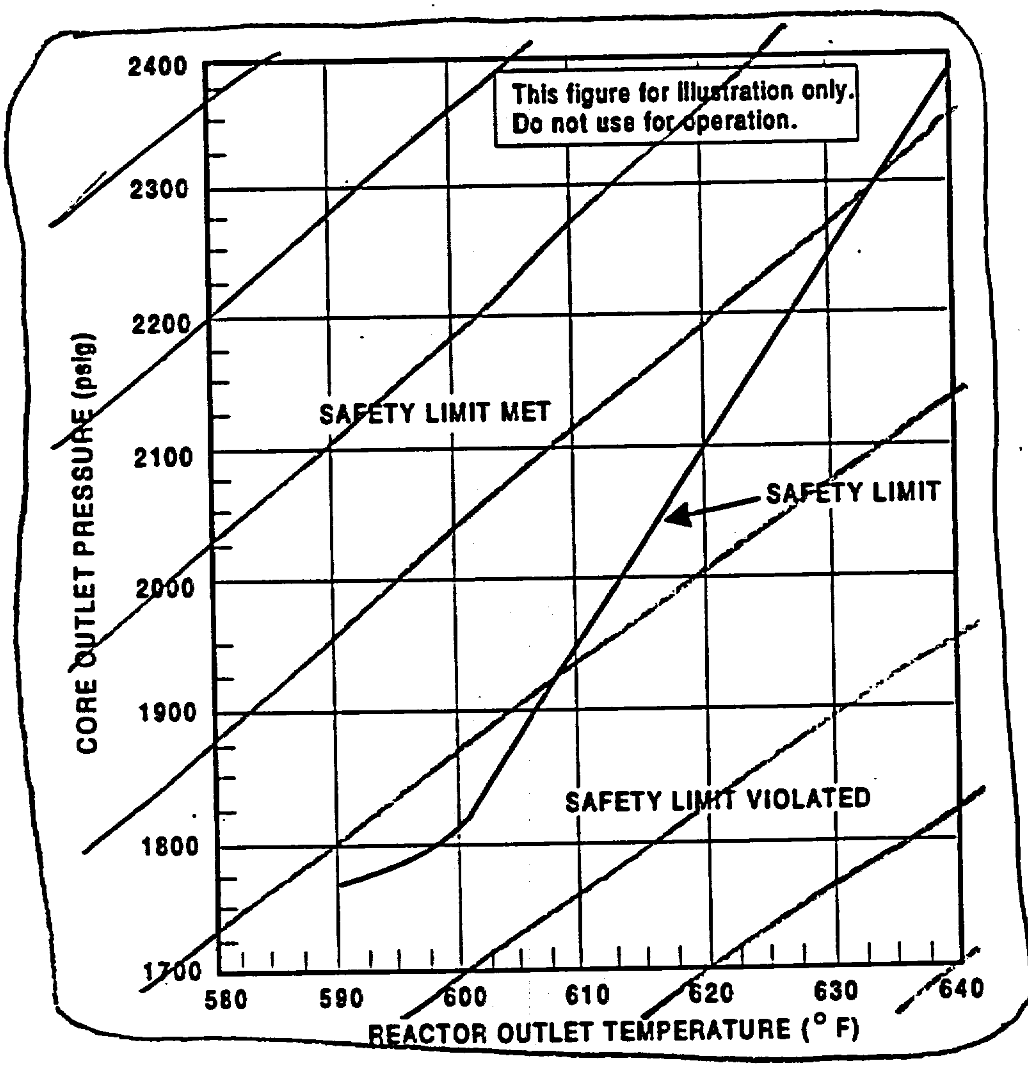
2.2.5 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72. 6.7.1.b

~~2.2.6 Within 24 hours, notify the [Vice President—Nuclear Operations].~~

~~2.2.7 Within 30 days, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC and the [Plant Superintendent, and Vice President—Nuclear Operations].~~

~~2.2.8 Operation of the plant shall not be resumed until authorized by the NRC.~~

(3)



19

Figure 2.1.1-1 (page 1 of 1)
Reactor Coolant System Departure from Nucleate Boiling Safety Limits

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that reactor core SLs ensure specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and ~~anticipated operational occurrences (AOOs)~~. abnormalities 15 This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that the fuel centerline temperature stays below the melting temperature.

INSERT B2.0-1A 7

INSERT B2.0-1B 20

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. 14 INSERT B2.0-1C Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding-water (zirconium-water) reaction may take place. edit This chemical reaction results in oxidation of the fuel cladding ~~to~~ a structurally weaker form. edit The oxidized cladding then exists in This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

(continued)

<INSERT B2.0-1A>

Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2 (Ref. 2) and BWC (Ref. 3) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady state operation, normal operational transients and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC).

<INSERT B2.0-1B>

The 95 percent confidence level that DNB will not occur is preserved by ensuring that the DNBR remains greater than the DNBR design limit based on the applicable CHF correlation for the core design. In the development of the applicable DNBR design limit (Ref. 4), uncertainties in the core state variables, power peaking factors, manufacturing-related parameters, and the CHF correlation may be statistically combined to determine a statistical DNBR design limit. This statistical design limit protects the respective CHF design limit. Additional retained thermal margin may also be applied to the statistical DNBR design limit to yield a higher thermal design limit for use in establishing DNB-based core safety and operating limits. In all cases, application of statistical DNB design methods preserves a 95 percent probability at a 95 percent confidence level that DNB will not occur.

<INSERT B2.0-1C>

The maximum fuel centerline temperatures are given by the relationships defined in SL 2.1.1.1 for the respective fuel designs and are dependent on whether the TACO2 (Ref. 5) or TACO3 (Ref. 6) analysis was utilized.

BASES

BACKGROUND (continued)

The proper functioning of the Reactor Protection System (RPS) ~~and main steam safety valves (MSSVs)~~ prevents violation of the reactor core SLs. 5

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs abnormalities. The reactor core SLs are established to preclude violation of the following fuel design criteria: 15

a. There must be at least 95% probability at a 95% confidence level (95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience fuel centerline melting.

The RPS setpoints (Ref. 7), in combination with all the LCOs, are designed to prevent any ~~anticipated~~ analyzed combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a ~~departure from nucleate boiling ratio~~ DNBR of less than the DNBR limit and preclude the existence of flow instabilities. edit edit edit

Automatic enforcement of these reactor core SLs is provided by the following:

a. RCS High Pressure trip;

b. RCS Low Pressure trip;

c. Nuclear Overpower trip;

d. RCS Variable Low Pressure trip (also known as Pressure Temperature trip); 6

e. Reactor Coolant Pump to Power trip;

f. Nuclear Overpower RCS Flow and Axial Power Imbalance (caps) trip; and edit

g. ~~MSSVs~~ RCS High Temperature trip 5

The SL represents a design requirement for establishing the RPS trip setpoints identified previously.

(continued)

BASES (continued)

SAFETY LIMITS

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature stays below the melting point, or the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or the exit quality is within the limits defined by the DNBR correlation. In addition, ~~SL 2.1.1.2~~ ~~shows~~ the COLR identifies the pressure/temperature operating region that keeps the reactor from reaching an SL when operating up to design power~~, and it defines the safe operating region from brittle fracture concerns~~. 19 edit 19

INSERT B2.0-3A

The SLs are preserved by monitoring the process variable AXIAL POWER IMBALANCE to ensure that the core operates within the fuel design criteria. AXIAL POWER IMBALANCE protective limits are ~~provided in the COLR.~~ The trip setpoints are derived by adjusting the measurement system independent AXIAL POWER IMBALANCE protective limit given in the COLR to allow for measurement system observability and instrumentation errors. 5

~~Operation within these limits is ensured by compliance with the AXIAL POWER IMBALANCE protective limits~~ preserved by their corresponding RPS setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," ~~as specified~~ and are provided in the COLR. The AXIAL POWER IMBALANCE protective limits are separate and distinct from the AXIAL POWER IMBALANCE operating limits defined by LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits." The AXIAL POWER IMBALANCE operating limits in LCO 3.2.3, also specified in the COLR, preserve initial conditions of the safety analyses but are not reactor core SLs. edit

APPLICABILITY

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. ~~The MSSVs, or~~ automatic protection actions~~,~~ serve to prevent RCS heatup to reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1. 5

(continued)

<INSERT B2.0-3A>

The COLR presents the most limiting condition of pressure/temperature combinations for all possible reactor coolant pump maximum THERMAL POWER combinations. Analyses have been performed which bound the three pump and two pump (one pump in each loop) allowed operating conditions based on the expected minimum flow rates and maximum ALLOWABLE THERMAL POWER for these operating conditions.

BASES

APPLICABILITY (continued)

In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1 and 2.2.2

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the requirement to go to MODE 3 places the plant in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the plant to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

2.2.5

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. ~~3~~ 8).

edit

~~2.2.6~~

~~If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.~~

~~2.2.7~~

~~If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 4). A copy of the report shall also be submitted to the senior~~

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(continued)

BASES

~~SAFETY LIMIT VIOLATIONS~~

~~2.2.7 (continued)~~

~~management of the nuclear plant, and the utility Vice President—Nuclear Operations.~~

~~2.2.8~~

~~If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.~~

3

REFERENCES

SAR, Section 1.4,

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

edit 7 14 20 edit

INSERT B2.0-5A

7. ~~2.~~ FSAR, ~~Section~~ Chapters ~~1.1~~ 3 & 14.

8. ~~3.~~ 10 CFR 50.72.

~~4. 10 CFR 50.73.~~

3

<INSERT B2.0-5A>

2. BAW-10000A, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," Babcock & Wilcox, Lynchburg, VA, May 1976 . 7

3. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," Babcock & Wilcox, Lynchburg, VA, April 1985.

4. BAW-10187P-A, "Statistical Core Design for B&W Designed 177 FA Plants," Babcock & Wilcox, Lynchburg, VA, March 1994 20

5. BAW-10141P-A, Rev. 1, "TACO2 Fuel Pin Performance Analysis," Babcock & Wilcox, Lynchburg, VA, June 1983.

6. BAW-10162P-A, "TACO3 Fuel Pin Thermal Analysis Code," Babcock & Wilcox, Lynchburg, VA, October 1989. 14

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

INSERT B2.0-6A

15

~~According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation nor during anticipated operational occurrences (AOOs).~~ GDC 28, "Reactivity Limits" (Ref. 1), specifies that reactivity accidents including rod ejection do not result in damage to the RCPB greater than limited local yielding.

abnormalities

The design pressure of the RCS is 2500 psig. During normal operation and AOOs, the RCS pressure is kept from exceeding the design pressure by more than 10% in order to remain in accordance with ~~Section III of the ASME Code (Ref. 2).~~ the design codes (Ref. 3 and 4). Hence, the safety limit is 2750 psig. To ensure system integrity, all RCS components ~~are~~ were hydrostatically tested at 125% of design pressure prior to initial operation, according to the ~~ASME Code~~ design code requirements. ~~Inservice operational hydrotesting at 100% of design pressure is also required whenever the reactor vessel head has been removed or if other pressure boundary joint alterations have occurred. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).~~

8

INSERT B2.0-6B

12

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, operating in conjunction with the Reactor Protection System trip settings, ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. ~~2~~ 3). The design basis transient that is most influential for establishing the required relief capacity, and hence the valve size requirements and lift settings, is a rod withdrawal event from low power. ~~During the transient, no control actions are assumed except that the safety valves on the secondary plant are assumed to open~~

9

(continued)

<INSERT B2.0-6A>

In SAR, Section 1.4 (Ref. 1), GDC 14, "Reactor Coolant Pressure Boundary (RCPB)," and GDC 15, "Reactor Coolant System Design", address RCPB design and protection, respectively. The ANO-1 discussion regarding how GDC 15 is accomplished states that analysis and evaluation of all normal and abnormal operating conditions and transients are integrally related to all RCS and associated systems design. SAR Chapter 14 (Ref. 2) lists these abnormal operating conditions and transients and terms them "abnormalities". In addition,

<INSERT B2.0-6B

Inservice leak testing at not less than 2155 psig is also required, prior to MODE 2, following any opening of the reactor coolant system in accordance with ASME code, Section XI; IWA-5000. When performed at the end of refueling outages, this leak test also satisfies the requirements of IWB-2500, Table IWB-2500-1; Category B-P Items B15.10, B15.20, B15.30, B15.40, B15.50, B15.60, and B15.70 for all Class I pressure retaining components (Ref. 5).

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.~~ 9

The Startup event (rod withdrawal at low power event) (Ref. 2) is ~~The overpressure protection analyses (Ref. 4) and the safety analyses (Ref. 5) are~~ performed using conservative assumptions relative to pressure control devices. 18

More specifically, no credit is taken for operation of the following:

a. Electromatic Relief Valve (ERV); ~~Pressurizer power operated relief valves (PORVs);~~ 11

b. Steam line turbine bypass valves;

c. Control system runback of reactor and turbine power; and

d. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under ~~USAS, Section B31.1~~ USAS B31.7 (Ref. ~~6~~ 4), is ~~120%~~ 110% of design pressure. ~~The most limiting of these two allowances is the 110% of design pressure;~~ therefore, the SL on maximum allowable RCS pressure is 2750 psig. 10

Overpressurization of the RCS can result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. ~~7~~ 6).

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES during overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized significantly. 16

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the RCS pressure SL.

2.2.3

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. ~~8~~ 6).

The allowed Completion Time of 1 hour is based on the importance of reducing power level to a MODE ~~of operation~~ where the potential for challenges to safety systems is minimized. Edit

2.2.4

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the RCS pressure SL in MODE 3, 4, or 5 is potentially more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

2.2.5

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. ~~8~~ 7). EDIT

(continued)

BASES

~~SAFETY LIMIT VIOLATIONS (continued)~~

~~**2.2.6**~~

~~If the RCS pressure SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.~~

~~**2.2.7**~~

~~If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC, in accordance with 10 CFR 50.73 (Ref. 9). A copy of the report shall also be provided to the senior management of the nuclear plant, and the utility Vice President—Nuclear Operations and the [offsite reviewers specified in Specification 5.2.2] ["Offsite Review and Audit"].~~

~~**2.2.8**~~

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

3

REFERENCES

1. SAR, Section 1.4, ~~10 CFR 50, Appendix A,~~ GDC 14, GDC 15, and GDC 28~~, 1988~~. EDIT.

3 ~~2~~. ASME Boiler and Pressure Vessel Code, Section III, 1965 – S67, Article NB-7000. 13

5 ~~3~~. ASME Boiler and Pressure Vessel Code, Section XI, Article IW-5000.

~~4. BAW-10043, May 1972.~~ Chapter 3 18

2 ~~5~~. ~~B~~SAR, ~~Section~~ Chapter 14.~~2~~.

4 ~~6~~. ~~ASME~~ USAS B31.7 ~~B31.1~~ Nuclear Power ~~Standard Code for Pressure~~ Piping, 1969. ~~1967~~. 10

(continued)

BASES

REFERENCES (continued)

6 ~~7~~. 10 CFR 100.

7 ~~8~~. 10 CFR 50.72.

~~9. 10 CFR 50.73.~~ 3

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours;

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)	Exceptions to this Specification are stated in the individual Specifications. LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3 and 4.
LCO 3.0.5	Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.
LCO 3.0.6	When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.15, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.
LCO 3.0.7	Test Exception LCOs 3.1.8 and 3.1.9 allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

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

SR 3.0.2

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

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

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

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3

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

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

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

(continued)

3.0 SR APPLICABILITY

SR 3.0.4	Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 through LCO 3.0.7 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specification.

BASES

LCO 3.0.2 (continued)

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. Reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

BASES

LCO 3.0.3 (continued)

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

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

a. The LCO is now met.

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

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

The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

BASES

LCO 3.0.3 (continued)

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.12, "Fuel Handling Area Ventilation System." LCO 3.7.12 has an Applicability of "During movement of irradiated fuel assemblies in the fuel handling area." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.12 are not met while in MODE 1, 2, 3, or 4, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.12 of "Suspend movement of irradiated fuel assemblies in the fuel handling area" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

BASES

LCO 3.0.4 (continued)

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

Exceptions to LCO 3.0.4 are stated in the individual Specifications. The exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability associated with operation in MODES 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO,

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LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate

BASES

LCO 3.0.5 (continued)

OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a

BASES

LCO 3.0.6 (continued)

supported system to be declared inoperable or directs entry in Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the remaining OPERABLE support systems are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operation is being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When a loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately address the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

BASES

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs 3.1.8 and 3.1.9 allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

B 3.0 SURVEILLANCE REQUIREMENT (SR)

BASES

SRs

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

SR 3.0.1

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

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

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

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

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

Unplanned events may satisfy the requirements for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

a. Emergency feedwater (EFW) pump turbine maintenance during refueling that requires testing at steam pressures > 800 psi. However, if other appropriate testing is satisfactorily completed, the EFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the EFW pump testing.

b. High pressure injection (HPI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers unit operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply.

BASES

SR 3.0.2 (continued)

These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is the Reactor Building Leakage Rate Testing Program.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an actioh usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

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

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

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

BASES

SR 3.0.3 (continued)

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

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

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

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

SR 3.0.4

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

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

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

BASES

SR 3.0.4 (continued)

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

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances. when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability associated with operation in MODES 1, 2, 3 or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

CTS DISCUSSION OF CHANGES

ITS Section 3.0: LCO and SR Applicability

ADMINISTRATIVE

A1 The designated change represents a non-technical, non-intent change to the Arkansas Nuclear One, Unit 1 Current Technical Specifications (CTS) made to make the ANO-1 Improved Technical Specifications (ITS) consistent with the Babcock and Wilcox (B&W) revised Standard Technical Specification (RSTS), NUREG-1430, Revision 1. This change does not alter the requirements of the CTS or RSTS. Examples of this type of change include: wording preference; convention adoption; editorial, numbering and formatting changes; and hierarchy structure.

A2 The ANO-1 CTS Bases will be administratively deleted in their entirety in favor of the NUREG-1430 Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 The NUREG-1430 Limiting Condition for Operation (LCO) 3.0.5 has been adopted for use in the ITS. This LCO establishes documented control of allowances for restoring equipment to service under administrative controls after it has been removed from service or declared inoperable to comply with ACTIONS. This LCO is consistent with the ANO-1 operating philosophy that allows for administrative control of equipment when necessary to establish operability of the component(s) although not strictly provided for by the CTS. Without this allowance, certain components and systems could not be returned to an OPERABLE status after having been declared inoperable.

A4 The ITS will state that "LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4" which is not explicitly stated in the ANO-1 CTS 3.0.3. However, these MODE limitations are consistent with the current application of CTS 3.0.3 and the ANO-1 CTS Bases for 3.0.3. The Bases for CTS 3.0.3 presently state that the requirements of CTS 3.0.3 "do not apply in COLD SHUTDOWN and REFUELING SHUTDOWN" which infers applicability to MODES 1, 2, 3, and 4.

A5 The NUREG-1430 LCO 3.0.7 has been adopted for use in the ITS. This LCO provides documented control for the implementation of special test exceptions provided by ITS such as LCO 3.1.8 and LCO 3.1.9. ITS LCO 3.0.7 eliminates the confusion which would otherwise exist as to which LCOs apply during the performance of a special test or operation. ITS LCO 3.0.7 is consistent with the intent of CTS special test exceptions. This change provides clarity only and, therefore is an administrative change.

A6 The NUREG-1430 LCO 3.0.1 has been adopted as presented in the ITS and is therefore substituted for CTS LCO 3.0.1. The ANO-1 CTS has been reviewed against the RSTS LCO and no technical or intent change exists with the adoption of the specification.

A7 The NUREG-1430 Limiting Condition for Operation (LCO) 3.0.2 has been adopted in its entirety and has been substituted for CTS LCO 3.0.2. The ANO-1 CTS has been reviewed against the RSTS LCO and no technical or intent change exists with the adoption of the RSTS. In association with the adoption of RSTS Specifications LCO 3.0.5 and LCO 3.0.6 which are unlike any CTS, specific provision has been provided for their exception in the ITS 3.0.2. This is an administrative change associated with the adoption of these Specifications in the ITS. (Reference DOC A3, above.)

A8 The NUREG-1430 Surveillance Requirement (SR) 3.0.1 has been adopted in the ITS. The RSTS SR presents requirements consistent with ANO-1 CTS 4.0.1 and the partial contents of CTS 4.0.3. The first sentence of ANO-1 CTS 4.0.3 (i.e. Failure to perform a SR within the allowed surveillance interval....) has been incorporated into ITS SR 3.0.1. The fourth sentence of ANO-1 CTS 4.0.3 (i.e. Surveillance Requirements do not have to be performed on inoperable equipment.) has been incorporated into ITS SR 3.0.1. This is an administrative change because RSTS SR 3.0.1 is composed of the requirements already contained in the ANO-1 CTS.

A9 The NUREG-1430 SR 3.0.2 has been adopted in the ITS. The first paragraph of RSTS SR 3.0.2 presents requirements consistent with ANO-1 CTS 4.0.2. The second and third paragraphs of RSTS SR 3.0.2 are not presented in an ANO-1 CTS. The adoption of the RSTS SR 3.0.2 results in the ITS SR 3.0.2 statement, "For Frequencies specified as 'once,' the above interval extension does not apply," which clarifies that the 1.25 times the interval specified in the Frequency does not apply to certain surveillances. The interval extension concept is intended to provide scheduling flexibility for repetitive performances, and if Surveillances or Required ACTIONS are not repetitive and have no "interval ... as measured from the previous performance" no extension should be applied. This statement precludes extending these SR performances, and technically is an additional restriction because the ANO-1 CTS allows the extension to apply to all SRs. However, these sentences are consistent with the operating philosophy of ANO-1; and therefore, the adoption of this SR is considered an administrative change.

TECHNICAL CHANGE – MORE RESTRICTIVE

M1 SR 3.0.3 will be adopted as presented in NUREG-1430. Both RSTS SR 3.0.3 and CTS 4.0.3 provide for an allowable delay of ACTIONS in the event a surveillance requirement is not performed. RSTS SR 3.0.3 differs from CTS Specification 4.0.3 in that SR 3.0.3 allows relaxation of the requirements of an LCO for the lesser of either the *surveillance frequency* or an allowed 24 hour delay, while CTS 4.0.3 allows a delay of the required ACTIONS for up to 24 hours when the *allowable outage time* is less than 24 hours. The adoption of RSTS SR 3.0.3 is more restrictive for those instances when both the surveillance frequency and allowable outage time associated with a specification are less than 24 hours. Under the guidance of SR 3.0.3, initiation of the required ACTIONS of the specification would be required sooner. SR 3.0.3 is being adopted to be consistent with NUREG -1430.

M2 The NUREG-1430 LCO 3.0.3 has been adopted as presented in the ITS and is therefore substituted for CTS LCO 3.0.3. The RSTS LCO has been reviewed against the ANO-1 CTS and found to represent a more restrictive requirement than in the CTS. The ANO-1 CTS 3.0.3 requires that the Unit be placed in Hot Standby within the next six (6) hours following the one (1) hour preparatory period for the initiation of actions. When THERMAL POWER is greater than 2% RTP, this CTS requirement would result in the Unit having to reduce neutron power as indicated on the power range channels below 2% RTP. The ANO-1 CTS 3.0.3 then requires that the Unit be placed in Hot Shutdown within the following six (6) hours which would require that the reactor be made 1% dK/K subcritical. The cumulative actions allow a total of 13 hours within which the reactor must be taken to a 1% dK/K subcritical condition.

When THERMAL POWER is greater than 5% RTP, the ITS LCO 3.0.3 requirements will require the Unit be placed in MODE 3 within the next six (6) hours following the one (1) hour preparatory period for the initiation of actions. This RSTS requirement would result in the unit having to reduce Keff to less than 0.99. The ITS LCO 3.0.3 requires that actions equivalent to those in the CTS 3.0.3 (items 1 and 2) be accomplished within a time period 6 hours shorter than that required in the CTS. ITS LCO 3.0.3 will also impose a requirement to be in MODE 4 within 13 hours of entry into LCO 3.0.3. This requirement does not presently exist in the CTS 3.0.3. The requirement to be in MODE 4 within 13 hours means that the Unit must be cooled from an allowable temperature of greater than 525°F to less than 280°F within a six hour time period. The ITS then requires that the Unit be cooled to less than or equal to 200°F within thirty seven (37) hours of entry into the LCO. The CTS 3.0.3 only requires that the Unit be cooled from Hot Shutdown conditions (1% dK/K shutdown with Tavg greater than 525°F) to Cold Shutdown conditions (1% dK/K shutdown with Tavg less than or equal to 200°F) within the subsequent 24 hours following establishment of Hot Shutdown conditions. While the allowable time for the transition from MODE 3 to MODE 5 remains as 24 hours, the requirement to enter MODE 4 within a specified time period of 6 hours represents a new and more restrictive requirement in the ITS.

CTS DISCUSSION OF CHANGES

TECHNICAL CHANGE – LESS RESTRICTIVE

L1 The NUREG-1430 LCO 3.0.4 has been adopted as presented in the ITS and is therefore substituted for CTS LCO 3.0.4. The ANO-1 CTS 3.0.4 presently states "this provision shall not prevent passage through or to REACTOR OPERATING CONDITIONS as required to comply with ACTION requirements." Because the ITS Specification adopted the RSTS Specification LCO 3.0.4 in its entirety, the new ITS Specification will state that the "specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit." Thus, ITS LCO 3.0.4 is considered to be less restrictive than the CTS 3.0.4 because the ITS will allow shutdown of the unit regardless of the ability to satisfy the requirements of lower MODE LCOs. The current interpretation of the CTS specification requires that the lower MODE LCOs be met for shutdown of the unit. The adoption of this less restrictive requirement is considered to be acceptable in that it is generally considered to be "safer" when the unit is subcritical, cooled down and depressurized. These lower MODE conditions generally constitute operating regions where the greatest margins of safety exist.

Associated with the adoption of STS LCO 3.0.4 is an administrative requirement that the individual specifications be evaluated to determine which LCOs should have specific restrictions on MODE changes or required ACTIONS. The results of this evaluation will be summarized in a matrix as specified by the STS Reviewer's Note. In addition, each Specification which requires specific limitations on MODE entry or Required ACTION compliance will be discussed with the Specification.

The NUREG-1430 SR 3.0.4 has also been adopted in the ITS. This SR incorporates requirements consistent with CTS 4.0.4. However, SR 3.0.4 also contains a provision such that "changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented." Thus, ITS SR 3.0.4 is considered to be less restrictive than the CTS 4.0.4 because the ITS will allow shutdown of the unit regardless of the ability to satisfy the requirements of lower MODE SRs. The current interpretation of the CTS specification requires that the lower MODE SRs be met for shutdown of the unit. The reasoning presented above for the incorporation of LCO 3.0.4 applies to this change as well.

L2 The incorporation of RSTS Specification LCO 3.0.6 represents less restrictive operating requirements for the facility. Specifically, the new specification provides clarification of the required ACTIONS when a support system is determined to be inoperable. Present operating philosophy and regulatory guidance would generally require "cascading" implementation of the ACTIONS for all supported systems or components. The incorporation of this Specification in the ITS resolves inconsistencies and ambiguities that exist between the ACTIONS of support systems and the ACTIONS of supported systems. Accompanying the incorporation of LCO 3.0.6 into the ITS will be the adoption of the Safety Function Determination Program (SFDP). This program ensures loss of safety function is detected and appropriate ACTIONS are taken when two or more LCOs are not met. Upon entry into LCO 3.0.6, an evaluation is required to be made to determine if a loss of safety function exists. If the SFDP identifies that a loss of safety function exists, the appropriate Conditions and Required ACTIONS of the LCO in which the loss of safety function exists are required to be entered. Although adoption of LCO 3.0.6 is less restrictive than the CTS requirements, the implementation of the Safety Function Determination Program will provide adequate measures to determine the existence of a loss of safety function and require entry into the Conditions and Required ACTIONS of the applicable LCO(s). The conversion of the CTS to ITS will generally include the adoption of Completion Times for Required ACTIONS which are consistent between supporting and supported components/systems (each occurrence will be individually evaluated). This standardization of Completion Times eliminates much of the ambiguity that exists between CTS Specifications. Since the practice of "cascading" presents an administrative burden on control room personnel with no real benefit to safety, this change represents an increase in plant safety as the control room personnel are allowed to concentrate on restoration of the system or component and are no longer distracted by the administrative burden of "cascading."

L3 SR 3.0.3 will be adopted as presented in NUREG-1430. The sentences "The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The time at which the ACTION is taken may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours." have been changed to "If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance." The three major differences are: 1) this change allows the requirement to declare the equipment inoperable (LCO not met) to be delayed for up to 24 hours instead of the LCO being declared not met at the time it is identified that the Surveillance has not been performed but delaying entry into the Required ACTIONS for up to 24 hours, 2) the change applies to all Required ACTIONS instead of only those Required ACTIONS whose allowable outage time limits (hereafter referred to as Completion Times consistent with the proposed ANO-1 ITS) are less than 24 hours, and 3) the delay is limited to the specified Frequency if the specified Frequency is less than 24 hours (See M1 above). The reasons for these changes are to prevent potential misuse of SR 3.0.3 and to provide for consistent application of the 24 hour delay regardless of the Completion Time for the associated Conditions. As stated in NRC Generic Letter 87-09, "It is overly conservative to assume that systems or components are inoperable when a surveillance has not been performed. The opposite is in fact the case, the vast majority of surveillances demonstrate that systems or components in fact are operable. When a Surveillance is missed, it is primarily a question of operability that has not been verified by the performance of the required surveillance." Based on consideration of plant conditions, adequate planning, availability of personnel, the time required to perform the Surveillance and the safety significance of the delay in completing the Surveillance, the NRC concluded, in Generic Letter 87-09, that 24 hours is an acceptable time limit for completing a missed Surveillance when the allowable outage times are less than the 24 hour limit or a shutdown is required to comply with Required ACTIONS, and that this 24 hour deferral should apply to all systems or components, regardless of the length of the Completion Time due to the overly conservative assumption that systems and components are inoperable when a surveillance has not been performed. Therefore, this change to LCO 4.0.3 (proposed SR 3.0.3) represents a technical enhancement.

CTS DISCUSSION OF CHANGES

LESS RESTRICTIVE – ADMINISTRATIVE DELETION OF REQUIREMENTS

LA None

ITS

~~3. LIMITING CONDITIONS FOR OPERATION~~ (LCO)

]A1

3.0 LIMITING CONDITION FOR OPERATION ~~(GENERAL)~~ APPLICABILITY

LCO 3.0.1

3.0.1 The Limiting Conditions for Operation requirements shall be applicable during the REACTOR OPERATING CONDITIONS or other conditions ~~specified~~ for each specification. A6

LCO 3.0.2

3.0.2 Adherence to the requirements of the Limiting Condition for Operation within the specified time interval shall constitute compliance with the specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, no further actions need be taken. A7

LCO 3.0.3

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated Action requirements, within one hour action shall be initiated to place the unit in an OPERATING CONDITION in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,

2. At least HOT SHUTDOWN within the following 6 hours, and

3. At least COLD SHUTDOWN within the subsequent 24 hours.

M2

Where corrective measures are completed that permit operation under the Action requirements, the Action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. ~~Exceptions to these requirements are stated in the individual Specifications.~~ LCO 3.03 is only applicable in MODES 1, 2, 3 and 4. A4

LCO 3.0.4

3.0.4 Entry into a REACTOR OPERATING CONDITION or other specified condition shall not be made when the conditions of the Limiting Conditions for Operation are not met and the associated action requires a shutdown if they are not met within a specified time interval. Entry into a REACTOR OPERATING CONDITION or other specified condition may be made in accordance with Action requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to REACTOR OPERATING CONDITIONS as required to comply with Action requirements. Exceptions to these requirements are stated in the individual specification. L1

<ADD LCO 3.0.5> A3

<ADD LCO 3.0.6> L2

<ADD LCO 3.0.7> A5

'3.0

ITS

~~LIMITING CONDITION FOR OPERATION (continued)~~ A1

~~3.0.5 When a system, subsystem, train, component or device is determined to b inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE; or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, within 2 hours action shall be initiated to place the unit in an OPERATING CONDITION in which the applicable Limiting Condition for Operation does not apply by placing it, as applicable, in:~~ — LATER

(LATER) (3.8)

~~1. At least HOT STANDBY within the next 6 hours,~~

~~2. At least HOT SHUTDOWN within the following 6 hours, and~~

~~3. At least COLD SHUTDOWN within the subsequent 24 hours.~~

~~This Specification is not applicable in Cold Shutdown or Refueling Shutdown.~~

~~BASES~~

~~3.0.1 through 3.0.4 Establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):~~

~~"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shutdown the reactor or follow any remedial Action permitted by the Technical Specification until the condition can be met."~~

~~3.0.1 Establishes the Applicability statement within each individual Specification as the requirement for when (i.e., in which operational modes or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The Action requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.~~ — A2

~~There are two basic types of Action requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the Action requirements. In this case, conformance to the Action requirements provides an acceptable level of safety for unlimited continued operation as long as the Action requirements continue to be met. The second type of Action requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to~~

BASES (continued)

restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these Actions are not completed withi the allowable outage time limits, a shutdown is required to place the facilit in a mode or condition in which the Specification no longer applies. It is not intended that the shutdown Action requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

The specified time limits of the Action requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the Action requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual Specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the Action requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with Action requirements, the plant may have entered a mode in which a new specification becomes applicable. In this case, the time limits of the Action requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Conditio for Operation are not met.

3.0.2 Establishes that noncompliance with a Specification exists when the requirements of the Limiting Condition for Operation are not met and the associated Action requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the Action requirements within the specified time interval constitutes compliance with a Specification and (2) completion of the remedia measures of the Action requirements is not required when compliance with a Limiting Condition for Operation is restored within the time interval specified in the associated Action requirements. A2

3.0.3 Establishes the shutdown Action requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated Action requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown mode when plant operation cannot be maintained within the limit for safe operation defined by the Limiting Conditions for Operation and its Action requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result i redundant systems or components being inoperable. One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower modes of operation permit the shutdown to proceed in a controlled and orderly

BASES (continued)

manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant systems and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the Action requirements are completed, the shutdown may be terminated. The time limits of the Action requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the Action requirements have been met or the time limits of the Action requirements have not expired, thus providing an allowance for the completion of the required Actions.

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in the COLD SHUTDOWN condition when a shutdown is required during the POWER mode of operation. If the plant is in a lower mode of operation when a shutdown is required, the time limit for reaching the next lower mode of operation applies. However, if a lower mode of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other applicable mode, is not reduced. For example, if HOT STANDBY is reached in 2 hours, the time allowed to reach HOT SHUTDOWN is the next 11 hours because the total time to reach HOT SHUTDOWN is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to POWER operation, a penalty is not incurred by having to reach a lower mode of operation in less than the total time allowed. A2

The same principle applies with regard to the allowable outage time limits of the Action requirements, if compliance with the Action requirements for one specification results in entry into a mode or condition of operation for another specification in which the requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may be added to the allowable outage time limits of the second specification. However, the allowable outage time limits of Action requirements for a higher mode of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower mode of operation.

The shutdown requirements of Specification 3.0.3 do not apply in COLD SHUTDOWN and REFUELING SHUTDOWN, because the Action requirements of individual specifications define the remedial measures to be taken.

3.0.4 Establishes limitations on mode changes when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher mode of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in a shutdown to comply with the Action requirements if a change in modes were permitted. The purpose of this specification is to ensure that facility operation is not

BASES (continued)

initiated or that higher modes of operation are not entered when corrective action is being taken to obtain compliance with a Specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with Action requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a mode change. Therefore, in this case, if the requirements for continued operation have been met in accordance with the requirements of the specification, then entry into that mode of operation is permissible. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

A2

When a shutdown is required to comply with Action requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower mode of operation. For the purpose of compliance with this specification the term 'shutdown' is defined as a required reduction in the REACTOR OPERATING CONDITION.

3.0.5 Delineates what additional conditions must be satisfied to permit operation to continue when a normal or emergency power source is not OPERABLE It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the Limiting Condition for Operation statements associated with individual systems, subsystems, trains, components or devices to be consistent with the Limiting Condition for Operation statements of the associated electrical power source. It allows operation to be governed by the time limits of the Limiting Condition for Operation for the normal or emergency power source, not the individual Limiting Condition for Operation statements for each system, subsystem, train component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specification 3.7.2.C provides for a 7 day out-of-service time when one emergency diesel generator is not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable Action statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to

ITS

SURVEILLANCE REQUIREMENTS (SR) APPLICABILITY A1

SR 3.0.1 4.0.1 Surveillance Requirements shall be met during the operational modes or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement. A8

SR 3.0.2 4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval. A9

SR 3.0.1 4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. A8

SR 3.0.3 The time limits of the Action requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The time at which the Action is taken may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the Action requirements are less than 24 hours. M1 L3

SR 3.0.1 Surveillance Requirements do not have to be performed on inoperable equipment. A8

SR 3.0.4 4.0.4 Entry into an operational mode or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to operational modes as required to comply with Action requirements. L1

<LATER> (5.0)

~~4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1,2, and 3 components shall be applicable as follows:~~

~~a. Inservice inspection of ASME Code Class 1,2, and 3 components and inservice testing of ASME Code Class 1,2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i),~~

LATER

ITS

<LATER>
(3.3A)
(3.3B)
(3.3C)
(3.3.D)

OPERATIONAL SAFETY ITEMS (continued)

4.1 (Continued)

b. ~~Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3.~~

c. ~~Discrepancies noted during surveillance testing will be corrected and recorded.~~

LATER

<LATER>
(3.2)

d. ~~A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system.~~

LATER

BASES

~~4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):~~

~~"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."~~

~~4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.~~

A2

BASES (continued)

4.0.2 Establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance intervals.

4.0.3 Establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the Action requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the Action requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the Action requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.2 was a violation of the OPERABILITY requirements of a Limiting Condition for Operation that is subject to enforcement action. Further, the failure to perform a surveillance within the provisions of Specification 4.0.2 is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10CFR 50.73(a)(2)(i)(B) because it is a condition prohibited by the plant's Technical Specifications. A2

If the allowable outage time limits of the Action requirements are less than 24 hours or a shutdown is required to comply with Action requirements, e.g., Specification 3.0.3, a 24-hour allowance is provided to permit a delay in implementing the Action requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown is required to comply with Action requirements or before other remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance

BASES (continued)

includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of mode changes imposed by Action requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.4 is allowed. If a surveillance is not completed within the 24-hour allowance, the time limits of the Action requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the Action requirements are applicable at the time that the surveillance is terminated. If the Action requirements are greater than 24 hours, sufficient time exists to complete the surveillance.

Surveillance Requirements do not have to be performed on inoperable equipment because the Action requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

A2

4.0.4 Establishes the requirement that all applicable surveillances must be met before entry into an operational mode or other condition of operation specified in the Specification. The purpose of this Specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a mode or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in operational modes or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with Action requirements, the provision of Specification 4.0.4 do not apply because this would delay placing the facility in a lower mode of operation.

4.0.5 Establishes the requirement that inservice inspection of ASME Code Class 1,2, and 3 components and inservice testing of ASME Code Class 1,2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. These requirements apply except when relief has been provided in writing by the Commission.

LATER

<LATER>
(5.0)

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 3.0: LCO and SR Applicability

Entergy Operations has evaluated these proposed Technical Specification changes and has determined that they involve no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as indicated below:

<u>3.0 L1</u>

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

The accident mitigation features of the plant are not affected by this proposed change. The change will not allow continuous operation such that a single failure will preclude the associated function from being performed. As required by the Reviewer's Note associated with RSTS SR 3.0.4, a review of the CTS will be conducted to determine where specific restrictions on MODE changes or Required Actions should be included in the individual LCOs. This review will take into account those accident mitigation features required to be in service in lower MODES of operation during a plant shutdown. This is acceptable as the components and systems associated with those accident mitigation features required in lower MODES of operation will continue to be required to be in service. The review required by the Reviewer's Note will ensure that allowing certain specific components to be inoperable during MODE reductions will not result in a significant increase in the probability or consequences of previously evaluated accidents.

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

The proposed change introduces no new mode of plant operation. Those accident mitigation features required to be operable in lower MODES of operation will still be required to be placed in service following implementation of this change. Only those components that have no safety significance in the lower MODES of operation will be affected by this change. The review required by the Reviewer's Note will ensure that allowing certain specific components to be inoperable during MODE reductions will not result in the creation of a new or different kind of accident.

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

This change may result in a net benefit to the margin of safety as it will allow the Operator to focus on those components that have been evaluated as being safety significant during a plant shutdown and reduce distractions that occur due to the operability requirements of non-safety significant components.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.0 L2

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

This proposed change does not affect previously analyzed events or any parameters associated with plant operations. The change will not allow continuous operation such that a single failure will preclude the associated function from being performed. This change deals only with the administrative burden associated with inoperable support systems. Upon discovery of an inoperable support system, a Condition will be entered that governs the restoration of the system or component within a timely manner. The accident mitigation features of the plant are not affected by the proposed change.

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

Because this change does not change the design configuration and introduces no new mode of plant operation, it will not create the possibility of a new or different kind of accident. This change deals only with the administrative burden associated with inoperable support systems. Upon discovery of an inoperable support system, a Condition will be entered that governs the restoration of the system or component within a timely manner.

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

This change may result in a net benefit to the margin of safety by reducing the Operator administrative burden associated with "cascading" TSs due to the inoperability of support systems. When implementing the CTS requirements for an inoperable support system, such as service water, the Operator must implement and track all of the LCO Actions and Allowable Outage Times (Completion Times) for the supported components. Following implementation of this change, the Operator will be required to enter the Condition for the support system and correct the Condition within the specified Completion Time. The supported components and systems will be evaluated under the Safety Function Determination Program for possible losses of safety function. This reduced administrative burden allows the Operator to concentrate on restoring the inoperable system or component, resulting in a net benefit to safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.0 L3

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

The proposed change incorporates the latest NRC guidelines relative to the performance of SRs, and does not result in changes to any hardware or operating methods. The Surveillance Frequencies are not assumed to be the initiator of any analyzed event. The change will not allow continuous operation such that a single failure will preclude the associated function from being performed. This change will allow delay in the entry into the ACTION Condition for up to 24 hours when a SR has not been performed within the requirements of SR 3.0.2. It is overly conservative to assume that systems or components are inoperable when a SR has not been performed. The opposite is in fact the case; the vast majority of SRs performed demonstrate that systems or components are OPERABLE. When a SR is not performed within the requirements of SR 3.0.2, it is primarily a question of OPERABILITY that has not been verified by the performance of the SR. Therefore, the consequences of an accident previously evaluated are not significantly increased since the most likely outcome of performing a Surveillance is that it does in fact demonstrate the system or component is OPERABLE.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated for ANO-1.

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

The delay in implementation of the ACTION Condition for the performance of a SR discovered to have not been performed within the requirements of SR 3.0.2 is acceptable based on the small probability of an event requiring the associated component and the Generic Letter 87-09 conclusion that "... the vast majority of surveillances demonstrate that systems or components are in fact operable." The requested allowance will provide sufficient time to perform the missed Surveillances in an orderly manner. Without the 24 hour delay, it is possible that the missed SR would force a plant shutdown, thus the plant could be shutting down while performing the missed SR. As a result of the delay, the potential for human error will be reduced. As such, any reduction in the margin of safety will be insignificant and offset by the benefit gained in plant safety due to avoidance of unnecessary plant transients and shutdowns.

ITS DISCUSSION OF DIFFERENCES
ITS Section 3.0: LCO and SR Applicability

1 NUREG LCO 3.0.1 - Incorporates TSTF-006, Rev. 1.

2 Not used.

3 Not used.

4 NUREG SR 3.0.2 - Incorporates TSTF-042.

5 NUREG SR 3.0.1 Bases - Incorporates TSTF-008, Rev. 2.

6 NUREG LCO 3.0.6 - Incorporates TSTF-166.

7 NUREG LCO 3.0.2 Bases - Incorporates TSTF-122.

8 NUREG SR 3.0.2 Bases – The Bases for SR 3.0.2 are revised to include reference to the Reactor Building Leakage Rate Testing Program consistent with current license basis. ANO-1 implemented a Reactor Building Leakage Testing Program as presented in CTS 4.4 and CTS 6.8.4. This program was implemented by CTS Amendment 185 dated October 3, 1996.

9 NUREG Bases SR 3.0.1 - Incorporates TSTF-043 except as follows: Included editorial changes to reflect unit specific system nomenclature.

10 NUREG LCO 3.0.3 Bases – Bases reference to LCO 3.7.14, "Fuel Storage Pool Water Level," was revised to refer to LCO 3.7.12, "Fuel Handling Area Ventilation System," because the fuel storage pool water level specification was not adopted in the ITS. This change is editorial in that revises the NUREG Bases to refer to a Specification that will actually be included in the ITS. This change neither adds new requirements nor removes any existing requirements but rather establishes an editorially correct reference within the ITS.

11 NUREG LCO 3.0.4 – Incorporates TSTF-104.

12 NUREG LCO 3.0.4 Bases – Incorporates TSTF-165.

13\. NUREG LCO 3.0.6 Bases - Incorporates TSTF-273, Rev 2.

CTS

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions ~~in the Applicability~~, except as provided in LCO 3.0.2 and LCO 3.0.7. 1 3.0.1

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6. 3.0.2

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in: 3.0.3

a. MODE 3 within 7 hours;

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This 3.0.4

(continued)

CTS

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued) — Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. — 3.0.4 cont.

Exceptions to this Specification are stated in the individual Specifications. ~~These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.~~ — 11

LCO ~~SR~~ 3.0.4 is only applicable for entry into a MODE ~~Mode~~ or other specified condition in the Applicability in MODES ~~Modes~~ 1, 2, 3 and 4. — Edit

~~[Reviewer's Note: LCO 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, LCO 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4. The MODE change restrictions in LCO 3.0.4 were previously applicable in all MODES. Before this version of LCO 3.0.4 can be implemented on a plant-specific basis, the licensee must review the existing technical specifications to determine where specific restrictions on MODE changes or Required Actions should be included in individual LCOs to justify this change; such an evaluation should be summarized in a matrix of all existing LCOs to facilitate NRC staff review of a conversion to the STS.]~~

LCO 3.0.5 — Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY. — N/A

(continued)

CTS

3.0 LCO APPLICABILITY (continued)

LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an ~~additional~~ evaluation~~s and limitations may be required in~~ shall be performed in accordance with Specification 5.5.15, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

N/A

6

edit

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

LCO 3.0.7

Test Exception LCOs ~~[3.1.9, 3.1.10, 3.1.11 and 3.4.19]~~ 3.1.8 and 3.1.9 allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

edit
N/A

CTS

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits. 4.0.1 4.0.3

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. 4.0.2

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

If a Required Action requires performance of a Surveillance or its Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. 4

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance. 4.0.3

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

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

(continued)

CTS

3.0 SR APPLICABILITY

SR 3.0.3 (continued)	declared not met, and the applicable Condition(s) must be entered.	
SR 3.0.4	Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4. ~~[Reviewer's Note: SR 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, SR 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4. The MODE change restrictions in SR 3.0.4 were previously applicable in all MODES. Before this version of SR 3.0.4 can be implemented on a plant-specific basis, the licensee must review the existing technical specifications to determine where specific restrictions on MODE changes or Required Actions should be included in individual LCOs to justify this change; such an evaluation should be summarized in a matrix of all existing LCOs to facilitate NRC staff review of a conversion to the STS.]~~	4.0.4

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO ~~3.0.6~~ 3.0.7 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

Edit

LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering

(continued)

BASES

LCO 3.0.2 (continued)

ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications. EDIT

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. Reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS ~~Alternatives that~~ would ~~not~~ result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time ~~other~~ conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed. 7

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.

(continued)

BASES (continued)

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

(continued)

BASES

LCO 3.0.3 (continued)

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

a. The LCO is now met.

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

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

The time limits of ~~Specification~~ LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed. Edit.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO ~~3.7.14, "Fuel Storage Pool Water Level."~~ 3.7.12, "Fuel Handling Area Ventilation System." LCO 3.7.~~14~~ 12 has an Applicability of "During movement of irradiated fuel 10

(continued)

BASES

the handling area

LCO 3.0.3 (continued)

assembles in the fuel ~~storage pool~~." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.14 are not met while in MODE 1, 2, 3, or 4, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.14 of "Suspend movement of irradiated fuel assemblies in the fuel ~~storage pool~~ handling area" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

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LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the

(continued)

BASES

LCO 3.0.4 (continued)

provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. INSERT B3.0-6A Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification. 11

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or ~~Mode~~ MODE 1 from ~~Mode~~ MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability associated with ~~only while~~ operating ~~ing~~ ON in MODES 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS ~~or~~ of individual specifications sufficiently define the remedial measures to be taken. CGP ~~[In some cases (e.g., ..) these ACTIONS provide a Note that states "While this LCO is not met, entry into a MODE or other specified condition in the Applicability is not permitted, unless required to comply with ACTIONS." This Note is a requirement explicitly precluding entry into a MODE or other specified condition of the Applicability.]~~ Edit edit

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with

(continued)

<INSERT B3.0-6A>

The exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time.

BASES

LCO 3.0.5 (continued)

the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

required testing

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the ~~allowed SRs~~. This Specification does not provide time to perform any other preventive or corrective maintenance.

required testing to demonstrate OPERABILITY

12

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.

Edit

An example of demonstrating the OPERABILITY of other equipment ~~being returned to service~~ is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of ~~an SR~~ on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of ~~an SR~~ on another channel in the same trip system.

required testing

12

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

(continued)

BASES

LCO 3.0.6 (continued)

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry in Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the remaining OPERABLE support systems are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and

(continued)

BASES

LCO 3.0.6 (continued) Required Actions of the LCO in which the loss of safety function exists are required to be entered. H 13

<INSERT B 3.0.9A> →

LCO 3.0.7 There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, ~~and to perform special evolutions.~~ Test Exception LCOs ~~[3.1.9, 3.1.10, 3.1.11, and 3.4.19]~~ 3.1.8 and 3.1.9 allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect. edit

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

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This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operation is being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When a loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately address the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

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

SR 3.0.1

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

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

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

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

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

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

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(continued)

<INSERT B3.0-10A>

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

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SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers ~~plant~~ unit operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

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The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is ~~a Surveillance with a Frequency of "in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions." The requirements of regulations take precedence over the TS. The TS cannot in and of themselves extend a test interval specified in the regulations.~~ the Reactor Building Leakage Rate Testing Program.

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(continued)

<INSERT B3.0-11A>

"Some examples of this process are:

a. Emergency feedwater (EFW) pump turbine maintenance during refueling that requires testing at steam pressures > 800 psi. However, if other appropriate testing is satisfactorily completed, the EFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the EFW pump testing.

b. High pressure injection (HPI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing."

BASES

SR 3.0.2 (continued)

~~Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."~~ (8)

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

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

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most

(continued)

BASES

SR 3.0.3 (continued)

probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance. SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

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

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

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

SR 3.0.4

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

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the

(continued)

BASES

SR 3.0.4 (continued)

failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

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

The provisions of SR 3.0.4 shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of $\boxed{\text{LCO}}$ SR 3.0.4 shall not prevent changes in MODEs cap or other specified conditions in the Applicability that result from any unit shutdown. — EDIT. — EDIT.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of

(continued)

BASES

SR 3.0.4 (continued)

the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability ~~only while operating~~ in MODES 1, 2, 3, or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

associated with operation

Edit.

This Section Addresses the Following Specifications:

NUREG-1430	ANO-1 ITS	Title
3.1.1	3.1.1	SHUTDOWN MARGIN
3.1.2	3.1.2	Reactivity Balance
3.1.3	3.1.3	Moderator Temperature Coefficient
3.1.4	3.1.4	CONTROL ROD Group Alignment Limits
3.1.5	3.1.5	Safety Rod Insertion Limits
3.1.6	3.1.6	AXIAL POWER SHAPING ROD (APSR) Alignment Limits
3.1.7	3.1.7	Position Indicator Channels
3.1.8	3.1.8	PHYSICS TEST Exceptions - MODE 1
3.1.9	3.1.9	PHYSICS TEST Exceptions - MODE 2

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 The SDM shall be within the limit provided in the COLR.

APPLICABILITY: MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM greater than or equal to the limit specified in the COLR.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Balance

LCO 3.1.2 The measured core reactivity balance shall be within ± 1% $\Delta k/k$ of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Measured core reactivity balance not within limit.	A.1 Re-evaluate core design and safety analysis and determine that the reactor core is acceptable for continued operation.	7 days
	<u>AND</u>	
	A.2 Establish appropriate operating restrictions and SRs.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	-----NOTES----- 1. The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. 2. This Surveillance is not required to be performed prior to entry into MODE 2. ---------- Verify measured core reactivity balance is within ± 1% $\Delta k/k$ of predicted values.	Once prior to entering MODE 1 after each fuel loading <u>AND</u> -----NOTE----- Only required after 60 EFPD ---------- 31 EFPD thereafter

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be non-positive whenever THERMAL POWER is $\geq$ 95% RTP and shall be less positive than 0.9×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is < 95% RTP.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within limits.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify MTC is within the limits.	Once prior to entering MODE 1 after each fuel loading

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 CONTROL ROD Group Alignment Limits

LCO 3.1.4 Each CONTROL ROD shall be OPERABLE and aligned to within 6.5% of its group average height.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CONTROL ROD inoperable, or not aligned to within 6.5% of its group average height, or both.	A.1.1 Verify SDM to be within the limit provided in the COLR.	1 hour AND Once per 12 hours thereafter
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2.1 Restore CONTROL ROD alignment.	2 hours
	OR	
	A.2.2.1 Reduce THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER.	2 hours
	AND	
	A.2.2.2 Verify the potential ejected rod worth is within the assumptions of the rod ejection analysis.	72 hours
	AND	
		(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2.3 ——NOTE—— Only required when THERMAL POWER is > 20% RTP. —————— Perform SR 3.2.5.1.	72 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3.	6 hours
C. More than one CONTROL ROD inoperable, or not aligned within 6.5% of its group average height, or both.	C.1.1 Verify SDM to be within the limit provided in the COLR.	1 hour
	OR	
	C.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	C.2 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual CONTROL ROD positions are within 6.5% of their group average height.	12 hours
SR 3.1.4.2	Verify CONTROL ROD freedom of movement for each individual CONTROL ROD that is not fully inserted.	92 days

SURVEILLANCE		FREQUENCY
SR 3.1.4.3	-----------------------------NOTE----------------------------- With rod drop times determined with at least one but less than four reactor coolant pumps operating, operation may proceed provided operation is restricted to the pump combination operating during the rod drop time determination or pump combinations providing less total reactor coolant flow. ---	
	Verify the rod drop time for each CONTROL ROD, from the fully withdrawn position, is ≤ 1.66 seconds from power interruption at the CONTROL ROD drive breakers to ¾ insertion (25% withdrawn position) with Tavg ≥ 525°F.	Once prior to reactor criticality after each removal of the reactor vessel head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Safety Rod Insertion Limits

LCO 3.1.5 Each safety rod shall be fully withdrawn.

-----NOTE-----

Not required for any safety rod inserted to perform SR 3.1.4.2.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One safety rod not fully withdrawn.	A.1.1 Verify SDM to be within the limit provided in the COLR. <u>OR</u> A.1.2 Initiate boration to restore SDM to within limit. <u>AND</u> A.2 Declare the rod inoperable.	1 hour 1 hour 1 hour
B. More than one safety rod not fully withdrawn.	B.1.1 Verify SDM to be within the limit provided in the COLR. <u>OR</u> B.1.2 Initiate boration to restore SDM to within limit. <u>AND</u> B.2 Be in MODE 3.	1 hour 1 hour 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each safety rod is fully withdrawn.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

LCO 3.1.6 Each APSR shall be OPERABLE and aligned to within 6.5% of its group average height.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One APSR inoperable, or not aligned to within 6.5% of its group average height, or both.	A.1 Restore APSR alignment. OR A.2 Reduce THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER	2 hours 2 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.6.1 Verify position of each APSR is within 6.5% of the group average height.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Position Indicator Channels

LCO 3.1.7 One position indicator channel for each CONTROL ROD and APSR shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each CONTROL ROD and APSR.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The required position indicator channel inoperable for one or more rods.	A.1 Declare the rod(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Perform CHANNEL CHECK of required position indicator channel.	12 hours
SR 3.1.7.2	Perform CHANNEL CALIBRATION of required position indicator channel.	18 months

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions - MODE 1

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.4, "CONTROL ROD Group Alignment Limits";
LCO 3.1.5, "Safety Rod Insertion Limits";
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits";
LCO 3.2.1, "Regulating Rod Insertion Limits," for the restricted operation region only;
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits";
LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits"; and
LCO 3.2.4, "QUADRANT POWER TILT (QPT)"

may be suspended, provided:

a. THERMAL POWER is maintained $\leq$ 85% RTP;

b. Nuclear overpower trip setpoint is $\leq$ 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP;

c. ------------------------------------NOTE------------------------------------
Only required when THERMAL POWER is > 20% RTP.

Linear Heat Rate (LHR) is maintained within the limits specified in the COLR; and

d. SDM is within the limits provided in the COLR.

APPLICABILITY: MODE 1 during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes
	<u>AND</u>	
	A.2 Suspend PHYSICS TESTS exceptions.	1 hour

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. THERMAL POWER > 85% RTP. OR Nuclear overpower trip setpoint > 10% higher than PHYSICS TESTS power level. OR Nuclear overpower trip setpoint > 90% RTP. OR ----------NOTE---------- Only required when THERMAL POWER is > 20% RTP. ------------------------------ LHR not within limits.	B.1 Suspend PHYSICS TESTS exceptions.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Verify THERMAL POWER is ≤ 85% RTP.	1 hour
SR 3.1.8.2	----------NOTE---------- Only required when THERMAL POWER is > 20% RTP. ------------------------------ Perform SR 3.2.5.1.	2 hours
SR 3.1.8.3	Verify nuclear overpower trip setpoint is ≤ 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP.	Prior to performance of PHYSICS TESTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.4	Verify SDM to be within the limits provided in the COLR.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.9 PHYSICS TESTS Exceptions - MODE 2

LCO 3.1.9 During performance of PHYSICS TESTS, the requirements of

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.4, "CONTROL ROD Group Alignment Limits";
LCO 3.1.5, "Safety Rod Insertion Limits";
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits";
LCO 3.2.1, "Regulating Rod Insertion Limits";
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended, provided:

a. THERMAL POWER is ≤ 5% RTP;

b. Nuclear overpower trip setpoint is set to ≤ 5% RTP;

c. Nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit is OPERABLE; and

d. SDM is within the limits provided in the COLR.

APPLICABILITY: During PHYSICS TESTS initiated in MODE 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER not within limit.	A.1 Open control rod drive trip breakers.	Immediately
B. SDM not within limit.	B.1 Initiate boration to restore SDM to within limit. AND B.2 Suspend PHYSICS TESTS exceptions.	15 minutes 1 hour

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Nuclear overpower trip setpoint is not within limit. <u>OR</u> Nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit inoperable.	C.1 Suspend PHYSICS TESTS exceptions.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.9.1	Verify THERMAL POWER is $\leq$ 5% RTP.	1 hour
SR 3.1.9.2	Verify nuclear overpower trip setpoint is $\leq$ 5% RTP.	Prior to performance of PHYSICS TESTS
SR 3.1.9.3	Verify SDM to be within the limit provided in the COLR.	24 hours

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions per GDC 26 (Ref. 1). In MODES 3, 4, and 5, SDM requirements provide sufficient reactivity margin to maintain the core subcritical during these conditions.

In MODES 1 and 2 while critical, SDM requirements are met by the worth of the withdrawn CONTROL RODS which provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and abnormalities. In MODE 2 while subcritical and in MODE 3, with all safety rods withdrawn and the RPS not in Shutdown Bypass, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all CONTROL RODS , assuming the single CONTROL ROD of highest reactivity worth is fully withdrawn. In MODES 3, 4, or 5, when all safety rods are not fully withdrawn or the RPS is in Shutdown Bypass, the SDM defines the degree of subcriticality required to be maintained, assuming the CONTROL ROD of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of CONTROL RODS and soluble boric acid in the Reactor Coolant System (RCS). In MODES 1 and 2, the CONTROL RODS can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, for analyzed events initiated in MODES 1 and 2, the CONTROL RODS, together with the Chemical Addition and Makeup and Purification System, provide SDM during power operation and are capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn (Ref. 1).

The Chemical Addition and Makeup and Purification System can compensate for fuel depletion, during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions (Ref. 1).

During operation in MODES 1 and 2, SDM control is ensured by operating with the safety rods fully withdrawn (LCO 3.1.5, "Safety Rod Insertion Limits") and the regulating rods within the limits of LCO 3.2.1, "Regulating Rod Insertion Limits." In MODE 3, consideration must be given to the position of the safety rods and whether the RPS is in Shutdown Bypass in determining the required SDM. When the unit is in MODES 3, 4, and 5, the SDM requirements are met by means of adjustments to the RCS boron concentration. Shutdown boron concentration requirements assume the highest worth rod is stuck in the fully withdrawn position to account for a postulated inoperable CONTROL ROD prior to reactor shutdown.

APPLICABLE SAFETY ANALYSES

For analyzed events in MODES 1 and 2 while critical, the minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and abnormalities, with assumption of the highest worth rod stuck out following a reactor trip.

In MODES 1 and 2 while critical, the acceptance criteria for SDM requirements are that specified acceptable fuel design limits are maintained. The SDM requirements must ensure that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events; and

b. The reactivity transients associated with postulated accident conditions are controllable with acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for abnormalities, and $\leq$ 280 cal/gm energy deposition for the rod ejection accident).

In MODES 3, 4, and 5, the SDM requirements must ensure that the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

In MODES 1 and 2 while critical, SDM satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3). In MODE 2 while subcritical and in MODES 3, 4, and 5, SDM satisfies Criterion 4 of 10 CFR 50.36.

LCO

In MODES 1 and 2, and in MODE 3 when all safety rods are fully withdrawn and the RPS is not in Shutdown Bypass, SDM is a core design condition that can be ensured through CONTROL ROD positioning (regulating and safety groups) and through the soluble boron concentration.

In MODE 3, when all safety rods are not fully withdrawn or the RPS is in Shutdown Bypass, and in MODES 4 and 5, SDM represents a required degree of subcriticality that assumes the highest reactivity worth CONTROL ROD is fully withdrawn.

APPLICABILITY

In MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to ensure that the reactor remains subcritical.

In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5 and LCO 3.2.1. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron source concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid addition tank (BAAT) or the borated water storage tank (BWST). The operator should borate with the best source available for the unit conditions.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

The SDM is verified by performing a reactivity balance calculation. The reactivity effects that are considered in the reactivity balance are dependent upon the operational MODE of the unit. In general, the reactivity balance includes the following reactivity effects:

a. RCS boron concentration;

b. CONTROL ROD position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration;

g. Isothermal temperature coefficient (ITC);

h. Moderator temperature coefficient (MTC); and

i. Doppler defect.

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical or critical but below the point of adding heat (POAH), and the fuel temperature will be changing at the same rate as the RCS.

Using the MTC and Doppler defect accounts for the reactivity effects of power operation above the POAH.

The Frequency of 24 hours is based on the generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which may include performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. SAR, Section 1.4, GDC 26.
2. SAR, Chapter 3.
3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Balance

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and abnormalities. Therefore, the reactivity balance is used as a measure of the agreement between the predicted core reactivity and the actual core reactivity during power operation. The periodic confirmation of the predicted core reactivity is necessary to ensure that safety analyses of design basis transients and accidents remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, CONTROL ROD, or burnable poison worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity. These could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing the predicted core reactivity with the actual core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations in ensuring the reactor can be brought safely to cold, subcritical conditions. The difference between the actual and predicted core reactivity is commonly referred to as a reactivity anomaly.

When the reactor is critical in MODE 1 or 2, a reactivity balance exists where the net reactivity is zero (referred to as the actual core reactivity state). A comparison of predicted core reactivity and the actual core reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions and the net reactivity is known to be zero. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as soluble boron and burnable absorbers, producing zero net reactivity.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel remaining from the previous cycle provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical, the excess positive reactivity of the fuel is compensated by burnable absorbers, CONTROL RODS, APSRs, thermal feedback from the fuel and moderator, fission product poisons (mainly xenon and samarium), epithermal energy neutron absorbers, neutron leakage and the reactor coolant system (RCS) boron concentration. During cycle operation, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the primary method of compensating for the reduction in excess reactivity is through a reduction in the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are the establishment of the reactivity balance limit to ensure that unit operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation is, therefore, dependent upon an accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as CONTROL ROD withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity (Ref. 2). These accident analysis evaluations rely on computer codes which have been qualified against available test data, operating unit data, and analytical benchmarks. Monitoring the core reactivity balance ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the requirements for reactivity control during the operating cycle.

The comparison between the actual reactivity condition of the critical reactor and the predicted initial core reactivity provides an opportunity for the normalization of the calculational models used to predict core reactivity. If the predicted core reactivity and the actual core reactivity at reference core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict reactivity requirements may not be accurate. If reasonable agreement between the actual and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the predicted reactivity condition from the actual reactivity condition during the operating cycle may be an indication that the calculational model is not adequate for the operating cycle or that an unexpected change in core conditions has occurred.

The normalization of the predicted reactivity parameters to the actual reactivity value is typically performed after reaching RTP following startup from a refueling outage, with the RCS temperature, CONTROL RODS, and APSRs in their reference positions and fission product poisons at their expected equilibrium concentrations. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated, as core conditions change during the cycle.

Reactivity balance satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled, once the core design is fixed. During operation, therefore, the conditions of the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the accident analyses are no longer valid, or that the uncertainties in the nuclear design methodology are larger than expected. A limit on the reactivity of ± 1% Δk/k has been established, based on engineering judgment. A ± 1% Δk/k deviation in the predicted reactivity from the actual reactivity condition of the reactor is larger than expected for normal operation and should therefore be evaluated.

When the predicted core reactivity is within 1% Δk/k of the actual reactivity value at steady state thermal conditions, the core is considered to be operating within acceptable design limits.

APPLICABILITY

In MODES 1 and 2, the limits on the core reactivity balance must be maintained to ensure an acceptable SDM and continued adherence to the assumptions used in the accident analyses. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed.

This Specification does not apply in MODES 3, 4, and 5, because the reactor is shutdown and the net reactivity condition of the reactor can not be easily determined and changes to core reactivity due to fuel depletion cannot occur.

In MODE 6, boron concentration requirements (LCO 3.9.1, "Refueling Boron Concentration") ensure that fuel movements are performed within acceptable bounds.

ACTIONS

A.1 and A.2

Should an anomaly develop between the actual core reactivity and the predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with the input assumptions used in the core design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of an abnormality or accident occurring during this period, and allows sufficient time to assess the physical condition of the core and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core reference conditions at the time of the reactivity balance, then a recalculation of the reactivity balance may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the appropriate reactivity parameter may be renormalized, and operation in MODE 1 may continue. If operational restrictions or additional surveillance requirements are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 7 days is adequate for preparing operating restrictions or surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity balance cannot be restored to within the ± 1% $\Delta k/k$ limit, the unit must be brought to a MODE in which the LCO does not apply. As a conservative measure, the unit must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then boration required by Required Action A.1 of LCO 3.1.1 would occur. The allowed Completion Time of 6 hours is reasonable, based on operating experience to reach the required unit conditions from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by a periodic reactivity balance calculation that compares the predicted core reactivity to the actual core reactivity condition (net reactivity of zero condition). The comparison is made considering that core conditions are fixed or stable, including CONTROL ROD and APSR positions, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed once prior to entering MODE 1 after each fuel loading as an initial check on core reactivity conditions and design calculations at BOC. A Note is included in the SR to indicate that the normalization of predicted core reactivity to the measured value may take place within the first 60 effective full power days (EFPD) after each fuel loading. The required Frequency of 31 EFPD, following the initial 60 EFPD after entering MODE 1 is acceptable, based on the slow rate of core reactivity changes due to fuel depletion and the presence of other indicators (QPT, etc.) for prompt indication of an anomaly. The 60 EFPD after entering MODE 1 allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. Another Note is included in the SRs to indicate that the performance of the Surveillance is not required for entry into MODE 2.

REFERENCES

1. SAR, Section 1.4, GDC 26, GDC 28, and GDC 29.
2. SAR, Chapter 3A and 14.
3. 10 CFR 50.36

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and associated Reactor Coolant System (RCS) shall be designed so that in the power operating range the net affect of the prompt inherent nuclear feedback characteristic tends to compensate for a rapid increase in reactivity.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). Therefore, with a negative MTC a coolant temperature increase will cause a reactivity decrease. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less than or equal to zero when THERMAL POWER is 95% RTP or greater. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional burnable absorbers to yield an MTC at BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles are evaluated to ensure the MTC does not become more negative than the value assumed in the safety analyses.

APPLICABLE SAFETY ANALYSES

Reference 2 contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are initial conditions in the safety analyses, and both values must be bounded. Values used in the analyses consider worst case conditions, such as very large soluble boron concentrations for overheating events, to ensure the accident results are bounding.

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis; and

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

Accidents that cause core overheating (either decreased heat removal or increased power production) must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the CONTROL ROD withdrawal transient from either zero or full THERMAL POWER. The limiting overheating event relative to plant response is based on the maximum difference between core power and steam generator heat removal during a transient. The most limiting event with respect to positive MTC is the startup accident.

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event that produces the most rapid cooldown of the RCS, and is therefore the most limiting event with respect to the negative MTC, is a steam line break (SLB) event. Following the reactor trip for the postulated EOC SLB event, the large moderator temperature reduction, combined with the large negative MTC, may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, a substantial fraction of core power may be produced with all CONTROL ROD assemblies inserted, except the most reactive one. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core power may be produced through the effects of subcritical neutron multiplication.

MTC values are bounded in reload safety evaluations, assuming steady state conditions at BOC and EOC.

In MODES 1 and 2 while critical, MTC satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3). In MODE 2 while subcritical, MTC satisfies Criterion 4 of 10 CFR 50.36.

LCO

LCO 3.1.3 requires the MTC to be within specified limits to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The LCO establishes a maximum positive value that can not be exceeded. The limit of +0.9E-4 $\Delta k/k/°F$ (corrected to 95% RTP) on positive MTC, when THERMAL POWER is < 95% RTP, ensures that core overheating accidents will not violate the accident analysis assumptions. The requirement for a non-positive MTC, when THERMAL POWER is ≥ 95% RTP, ensures that core operation will be stable.

MTC is a core physics parameter determined by the fuel and fuel cycle design and cannot be controlled directly once the core design is fixed during operation, therefore, the LCO can only be ensured through measurement. The surveillance check at BOC on MTC provides confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met.

APPLICABILITY

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from power operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure that startup and subcritical accidents, such as the uncontrolled CONTROL ROD or group withdrawal, will not violate the assumptions of the accident analysis. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES. However, the variation of MTC with temperature in MODES 3, 4, and 5 for DBAs initiated in MODES 1 and 2 is accounted for in the subject accident analysis. The variation of MTC with temperature assumed in the safety analysis, is accepted as valid once the BOC measurement is used for normalization.

ACTIONS

A.1

MTC is a core physics parameter determined by the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis assumptions. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, for reaching MODE 3 conditions from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The SR for measurement of the MTC at the beginning of each fuel cycle provides for confirmation of the limiting MTC values. The MTC changes slowly from most positive (least negative) to most negative value during fuel cycle operation, as the RCS boron concentration is reduced with fuel depletion.

The requirement for measurement, prior to initial operation in MODE 1, satisfies the confirmatory check on the most positive (least negative) MTC value. MTC values are extrapolated and compensated to permit direct comparison to the specified MTC limits.

REFERENCES

1. SAR, Section 1.4, GDC 11.

2. SAR, Chapter 3A and 14.

3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 CONTROL ROD Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY of the CONTROL RODS is an initial condition assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial condition assumption in the safety analysis that directly affects core power distributions and assumptions of SDM.

The applicable criteria for these design requirements are GDC 10, "Reactor Design," and GDC 26, "Reactivity Control System Redundancy-and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a CONTROL ROD to become inoperable or to become misaligned from its group. CONTROL ROD inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available CONTROL ROD worth for reactor shutdown. Therefore, CONTROL ROD alignment and OPERABILITY are related to core operation within design power peaking limits and the core design requirement of a minimum SDM.

Limits on CONTROL ROD alignment and OPERABILITY have been established, and all CONTROL ROD positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

CONTROL RODS are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its rod 3/4 inch for one revolution of the leadscrew, but at varying rates depending on the signal output from the Control Rod Drive Control System (CRDCS).

The CONTROL RODS are arranged into rod groups that are radially symmetric. Therefore, movement of the CONTROL RODS does not introduce radial asymmetries in the core power distribution. The CONTROL RODS provide required negative reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating rods provide reactivity control during normal operation and transients, and their movement is normally controlled in automatic by a rod control system.

The axial position of the CONTROL RODS is indicated by three independent systems, which are the relative position indicators, the absolute position indicators, and the zone reference indicators (see LCO 3.1.7, "Position Indicator Channels").

The relative position indicator transducer is a potentiometer that is driven by electrical pulses from the CRDCS. There is one counter for each CONTROL ROD drive. Individual rods in a group, when aligned to the same power supply, all receive the same signal to move; therefore, the counters for all rods in a group should normally indicate the same position. The Relative Position Indicator System is considered highly precise. However, if a rod does not move for each demand pulse, the counter will still count the pulse and incorrectly reflect the position of the rod.

The Absolute Position Indicator System provides a highly accurate indication of actual CONTROL ROD position, but at a lower precision than the relative position indicators. This system is based on the signals from a series of reed switches spaced along a tube.

Other reed switches included in the same tube with the absolute position indicator matrix provide full in and full out limit indications and position indications at 0%, 25%, 50%, 75%, and 100% travel. This series of seven indicators are called zone reference indicators.

APPLICABLE SAFETY ANALYSES

CONTROL ROD misalignment and inoperability accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing CONTROL ROD inoperability or misalignment are that:

a. There shall be no violations of:

   1. specified acceptable fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core must remain subcritical after an abnormality or accident.

Two types of misalignment are distinguished during MODES 1 and 2. During movement of a CONTROL ROD group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs when one CONTROL ROD drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return towards the original power due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local linear heat rates (LHRs).

The accident analysis and reload safety evaluations define regulating rod insertion limits that ensure the required SDM can always be achieved if the maximum worth CONTROL ROD is stuck fully withdrawn (Ref. 3). If a CONTROL ROD is stuck in or dropped in, continued operation is permitted if the increase in local LHR is within the design limits. The Required Action statements in the LCOs provide conservative reductions in THERMAL POWER and verification of SDM to ensure continued operation remains within the bounds of the safety analysis (Ref. 3).

Continued operation of the reactor with a misaligned or dropped CONTROL ROD is allowed if the local core LHRs are verified to be within their limits in the COLR. When a CONTROL ROD is misaligned, the assumptions that are used to determine the regulating rod insertion limits, APSR insertion limits, AXIAL POWER IMBALANCE limits, and QPT limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and local core LHRs must be verified directly by incore mapping. Bases Section 3.2, "Power Distribution Limits," contains a more complete discussion of the relation of LHR to the operating limits.

In MODES 1 and 2 while critical, the CONTROL ROD group alignment limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 4). In MODE 2 while subcritical, the CONTROL ROD group alignment limits satisfy Criterion 4 of 10 CFR 50.36.

LCO

The limits on CONTROL ROD group alignment, safety rod insertion, and APSR alignment, together with the limits on regulating rod insertion, APSR insertion, AXIAL POWER IMBALANCE, and QPT, ensure the reactor will operate within the fuel design criteria. The Required Actions in these LCOs ensure that deviations from the alignment limits will either be corrected or that THERMAL POWER will be adjusted, so that excessive local LHRs will not occur and the requirements on SDM and ejected rod worth are preserved.

The limit for individual CONTROL ROD misalignment is 6.5% (approximately 9 inches) deviation from the group average position. This value is established, based on the distance between reed switches, with additional allowances for uncertainty in the absolute position indicator amplifiers, group average position calculator, and asymmetric alarm or fault detector outputs. Therefore, no additional uncertainties are required to be incorporated in the implementing procedures.

For the purpose of complying with this LCO, the position of a misaligned rod is not included in the calculation of the rod group average position.

Failure to meet the requirements of this LCO may produce unacceptable LHRs, or unacceptable SDM or ejected rod worth, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on CONTROL ROD OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which significant neutron (or fission) power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and resultant local power peaking would not exceed fuel design limits. In MODES 3, 4, 5, and 6, the OPERABILITY of the CONTROL RODS has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during MODE 6.

ACTIONS

A.1.1

Compliance with Required Actions of Condition A allows for continued power operation with one CONTROL ROD inoperable, or misaligned from its group average position, or both. Since the rod may be inserted farther than the group average insertion for a long time, SDM must be evaluated. Ensuring the SDM meets the minimum requirement established in the COLR within 1 hour is adequate to determine that further degradation of the SDM is not occurring.

A.1.2

If the SDM is less than the limit specified in the COLR, then the restoration of the required SDM requires increasing the RCS boron concentration, since the CONTROL ROD may remain misaligned and not be providing its normal negative reactivity on tripping. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour to initiate boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

A.2.1

Alignment of the inoperable or misaligned CONTROL ROD may be accomplished by either moving the single CONTROL ROD to the group average position, or by moving the remainder of the group to the position of the single inoperable or misaligned CONTROL ROD. Either action can be used to restore the CONTROL RODS to a radially symmetric pattern. However, this must be done without violating the CONTROL ROD group sequence, overlap, and insertion limits of LCO 3.2.1, "Regulating Rod Insertion Limits," given in the COLR. THERMAL POWER must also be restricted, as necessary, to the value allowed by the insertion limits of LCO 3.2.1. The required Completion Time of 2 hours is acceptable because local xenon redistribution during this short interval will not cause a significant increase in LHR. This option of inserting the group to the position of the misaligned rod is not available if a safety rod is misaligned, since the limits of LCO 3.1.5, "Safety Rod

Insertion Limits," would be violated. If realignment of the CONTROL ROD to the group average or alignment of the group to the misaligned CONTROL ROD is not completed within 1 hour, the rod shall be considered inoperable.

A.2.2.1

Reduction of THERMAL POWER to ≤ 60% ALLOWABLE THERMAL POWER ensures that local LHR increases, due to a misaligned rod, will not cause the core design criteria to be exceeded. The required Completion Time of 2 hours allows the operator sufficient time for reducing THERMAL POWER.

A.2.2.2

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.65% Δk/k at RTP or 1.00% Δk/k at zero power (Ref. 3). This evaluation may require a computer calculation of the maximum ejected rod worth based on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the duration of time that operation is expected to continue with a misaligned rod. Should fuel cycle conditions at some later time become more bounding than those at the time of the rod misalignment, additional evaluation will be required to verify the continued acceptability of operation. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation.

A.2.2.3

Performance of SR 3.2.5.1 provides a determination of the local core LHRs using the Incore Detector System. Verification of the local core LHRs from an incore power distribution map is necessary to ensure that excessive local LHRs will not occur due to CONTROL ROD misalignment. This is necessary because the assumption that all CONTROL RODS are aligned (used to determine the regulating rod insertion, AXIAL POWER IMBALANCE, and QPT limits) is not valid when the CONTROL RODS are not aligned. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and adequate time is allowed to obtain an incore power distribution map.

Required Action A.2.2.3 is modified by a Note that requires the performance of SR 3.2.5.1 only when THERMAL POWER is greater than 20% RTP. This establishes a Required Action that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

B.1

If the Required Actions and associated Completion Times for Condition A are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

C.1.1

More than one CONTROL ROD becoming inoperable or misaligned from their group average position, or both, is not expected and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour allows the operator adequate time to determine the SDM.

C.1.2

If the SDM is less than the limit specified in the COLR, then the restoration of the required SDM requires increasing the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

C.2

If more than one CONTROL ROD is inoperable or misaligned from their group average position, continued operation of the reactor may cause the misalignment to increase, as the regulating rods insert or withdraw to control reactivity. If the CONTROL ROD misalignment increases, local power peaking may also increase, and local LHRs will also increase if the reactor continues operation at THERMAL POWER. The SDM is decreased when one or more CONTROL RODS become inoperable at a given THERMAL POWER level, or if one or more CONTROL RODS become misaligned by insertion from the group average position.

Therefore, it is prudent to place the reactor in MODE 3. LCO 3.1.4 does not apply in MODE 3 since excessive power peaking cannot occur. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual CONTROL RODS are aligned within 6.5% of their group average height limits at a 12 hour Frequency allows the operator to detect a rod that is beginning to deviate from its expected position. The specified Frequency takes into account other CONTROL ROD position information that is continuously available to the operator in the control room, so that during actual CONTROL ROD motion, deviations can immediately be detected.

SR 3.1.4.2

Verifying each CONTROL ROD is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each CONTROL ROD could result in radial tilts. Exercising each individual CONTROL ROD every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each CONTROL ROD enough to verify freedom of movement will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between typical performances of SR 3.1.4.2 (determination of CONTROL ROD OPERABILITY by movement), if a CONTROL ROD(S) is discovered to be immovable, but is otherwise determined to be capable of being fully inserted, the CONTROL ROD(S) may continue to be considered OPERABLE unless inoperable for some other reason. At any time, if a CONTROL ROD(S) is immovable, a determination of the capability to fully insert (OPERABILITY) the CONTROL ROD(S) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of CONTROL ROD drop time allows the operator to determine that the maximum CONTROL ROD drop time permitted is consistent with the assumed CONTROL ROD drop time used in the safety analysis. The CONTROL ROD drop time given in the safety analysis is 1.66 seconds to 3/4 position insertion (Ref. 5). This 1.66 seconds includes 0.14 seconds delay time for opening of the CRD breakers and for CRDM unlatch. Using the CONTROL ROD position versus time and time versus reactivity insertion curves gives a value of 1.4 seconds to 2/3 reactivity insertion upon which the accident analysis is based (Ref. 3). The former value is used in the Surveillance because the zone reference lights are located at 25% insertion intervals. The zone reference lights will activate at 3/4 insertion to give an indication of the CONTROL ROD drop time and CONTROL ROD location. The CONTROL ROD drop time is the total elapsed time from the loss of power to the control rod drive (CRD) breaker under voltage coils until the CONTROL ROD has completed approximately 104 inches of travel from the fully withdrawn position. The safety analysis has included a CRD breaker time delay of 0.080 seconds in SAR Chapter 14 (Ref. 3). If the trip test measurement is begun with the opening of the CRD breakers, the required trip insertion time shall be reduced to 1.58 seconds and the CRD breaker time delay shall be verified to be less than or equal to 0.080 seconds.

Measuring CONTROL ROD drop times, prior to reactor criticality after reactor vessel head removal, ensures that the reactor internals and CRDM will not interfere with CONTROL ROD motion or CONTROL ROD drop time. This Surveillance is performed during a unit outage, due to the unit conditions needed to perform the SR and the potential for an unplanned unit transient if the Surveillance were performed with the reactor at power.

This testing is normally performed with all reactor coolant pumps operating and average moderator temperature ≥ 525°F to simulate a reactor trip under actual conditions. However, if the CONTROL ROD drop times are determined with less than four reactor coolant pumps operating, a Note allows operation to continue, provided operation is restricted to the pump combination utilized during the CONTROL ROD drop time determination or pump combinations providing less total reactor coolant flow.

REFERENCES

1. SAR, Section 1.4, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. SAR, Chapter 3A and 14.
4. 10 CFR 50.36.
5. SAR, Chapter 3.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Safety Rod Insertion Limit

BASES

BACKGROUND

The insertion limits of the CONTROL RODS are initial condition assumptions in all safety analyses that assume CONTROL ROD insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected rod worth, and initial reactivity insertion rate.

The applicable criteria for the reactivity and power distribution design requirements are GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on safety rod insertion have been established, and all CONTROL ROD positions are monitored and controlled during operation in MODES 1 and 2 to ensure that the reactivity limits, ejected rod worth, and SDM limits are preserved.

The regulating groups are used for precise reactivity control of the reactor. The positions of the regulating groups are normally automatically controlled by the automatic control system, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). In MODES 1 and 2, the regulating groups must be maintained above designated insertion limits and are typically near the fully withdrawn position during normal operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature and fuel burnup.

The safety groups can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The safety groups are controlled manually by the control room operator. Prior to entry into MODE 2 from MODE 3, the safety groups must be fully withdrawn. The safety groups must be completely withdrawn from the core prior to withdrawing any regulating groups during an approach to criticality. The safety groups remain in the fully withdrawn position until the reactor is shut down. They add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

On a reactor trip, all CONTROL RODS, except the most reactive rod, are assumed to insert into the core. The safety groups shall be at their fully withdrawn limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The regulating groups may be partially inserted in the core as allowed by LCO 3.2.1, "Regulating Rod Insertion Limits." The safety group and regulating rod group insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from RTP. The combination of regulating groups and safety groups (less the most reactive rod, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power and to achieve the required SDM at rated no load temperature (Ref. 3).

The acceptance criteria for addressing safety and regulating rod group insertion limits and inoperability or misalignment are that:

a. There shall be no violations of:

   1. specified acceptable fuel design limits, or
   2. RCS pressure boundary integrity; and

b. The core must remain subcritical after an abnormality. Although the SAR does not state this as an acceptance criteria for the main steam line break event, B & W has placed a design objective on this event that the core remains subcritical throughout the event (Ref. 4).

In MODES 1 and 2 while critical, the safety rod insertion limits satisfy Criteria 2 and 3 of 10 CFR 50.36 (Ref. 5). In MODE 2 while subcritical, the safety rod insertion limits satisfy Criterion 4 of 10 CFR 50.36.

LCO

The safety groups must be fully withdrawn any time the reactor is in MODE 1 or 2. This LCO in combination with LCO 3.2.1 ensures that a sufficient amount of negative reactivity is available to shut down the reactor and achieve the required SDM following a reactor trip.

This LCO has been modified by a Note indicating the LCO requirement is suspended for those safety rods which are inserted solely due to testing in accordance with SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the safety group to move below the LCO limits, which would normally violate the LCO.

APPLICABILITY

The safety groups must be within their insertion limits with the reactor in MODES 1 and 2. This LCO in combination with LCO 3.2.1 ensures that a sufficient amount of negative reactivity is available to shut down the reactor and achieve the required SDM following a reactor trip. Refer to LCO 3.1.1 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

ACTIONS

A.1.1, A.1.2, and A.2

The safety rod must be declared inoperable within a 1 hour time frame. This requires entry into LCO 3.1.4, "CONTROL ROD Group Alignment Limits." In addition, since the safety rod may be inserted farther than the group average insertion for a long time, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour is adequate to determine that further degradation of the SDM is not occurring.

Restoration of the required SDM, if necessary, requires increasing the boron concentration, since the safety rod may remain misaligned and not be providing its normal negative reactivity on tripping. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

The allowed Completion Time of 1 hour provides an acceptable time for evaluating and repairing minor problems without allowing the unit to remain in an unacceptable condition for an extended period of time.

B.1.1 and B.1.2

When more than one safety rod is not fully withdrawn, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration to restore SDM.

In this situation, SDM verification must include the worth of any rod not capable of being fully inserted as well as the CONTROL ROD of maximum worth.

B.2

If more than one safety rod is not fully withdrawn, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that each safety rod is fully withdrawn ensures the safety rods are available to provide reactor shutdown capability.

Verification that individual safety rod positions are fully withdrawn at a 12 hour Frequency allows the operator to detect a safety rod beginning to deviate from its expected position. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of the safety rods.

REFERENCES

1. SAR, Section 1.4, GDC 10, GDC 26, and GDC 28.
2. 10 CFR 50.46.
3. SAR, Chapters 3 and 4.
4. BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Rev. 2.
5. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

BASES

BACKGROUND

The OPERABILITY of the APSRs and APSR alignment are initial condition assumptions in the safety analysis that directly affect core power distributions. The applicable criteria for these power distribution design requirements are GDC 10, "Reactor Design," and GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Mechanical or electrical failures may cause an APSR to become inoperable or to become misaligned from its group. APSR inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution. Therefore, APSR alignment and OPERABILITY are related to core operation within design power peaking limits.

Limits on APSR alignment and OPERABILITY have been established, and all APSR and CONTROL ROD positions are monitored and controlled during power operation to ensure that the power distribution limits defined by the design peaking limits are preserved.

APSRs are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its rod 3/4 inch for one revolution of the leadscrew, but at varying rates depending on the signal output from the Control Rod Drive Control System (CRDCS).

The APSRs are arranged into groups that are radially symmetric. Therefore, movement of the APSRs does not introduce radial asymmetries in the core power distribution. The APSRs, which are used to assist in control of the axial power distribution, are positioned manually and do not trip.

LCO 3.1.6 is conservatively based on use of black (Ag-In-Cd) APSRs and bounds use of gray (Inconel) APSRs. The reactivity worth of black APSRs is greater than that of gray APSRs; thus the impact of black APSR misalignment on the core power distribution is greater.

APPLICABLE SAFETY ANALYSES

There are no explicit safety analyses associated with misaligned APSRs. However, alignment of the APSRs is required to prevent inducing a QUADRANT POWER TILT. The LCOs governing APSR alignment are provided because the power distribution analysis supporting LCO 3.2.1, LCO 3.2.3 and LCO 3.2.4 assumes the APSRs are aligned.

During movement of an APSR group, one rod may stop moving while the other rods in the group continue. This condition may cause excessive power peaking. Continued operation of the reactor with a misaligned APSR is allowed if Section 3.2, "Power Distribution Limits," are preserved.

Because ANO-1 uses gray APSRs, the APSR alignment limits satisfy Criterion 4 of 10 CFR 50.36 (Ref. 3).

LCO

The limits on CONTROL ROD group alignment, safety rod withdrawal, and APSR alignment, together with the limits on regulating rod insertion, APSR insertion, AXIAL POWER IMBALANCE, and QPT, ensure the reactor will operate within the fuel design criteria. The Required Action in this LCO ensures deviations from the alignment limits will be adjusted so that excessive local LHRs will not occur.

The limit for individual APSR misalignment is 6.5% (approximately 9 inches) deviation from the group average position. This value is established based on the distance between reed switches, with additional allowances for uncertainty in the absolute position indicator amplifiers, group average position calculator, and asymmetric alarm or fault detector outputs. Therefore, no additional uncertainties are required to be incorporated in the implementing procedures. The position of an inoperable APSR is not included in the calculation of the APSR group's average position.

Failure to meet the requirements of this LCO may produce unacceptable LHRs, which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on APSR OPERABILITY and alignment are applicable in MODES 1 and 2, because these are the only MODES in which significant neutron (or fission) power is generated, and the OPERABILITY and alignment of APSRs have the potential to affect the safety of the unit. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down, and excessive local LHRs cannot occur from APSR misalignment.

ACTIONS

The ACTIONS described below are required if one APSR is inoperable. The unit is not allowed to operate with more than one inoperable APSR. This would require the reactor to be placed in MODE 3, in accordance with LCO 3.0.3.

A.1

An alternate to realigning a single misaligned APSR to the group average position is to align the remainder of the APSR group to the position of the misaligned or inoperable APSR, while maintaining APSR insertion, in accordance with the limits in the COLR. This restores the alignment requirements. Deviations up to 2 hours will not cause significant xenon redistribution to occur. This alternative assumes the APSR group movement does not cause the limits of LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," to be exceeded. For this reason, APSR group movement is only practical for instances where small movements of the APSR group are sufficient to re-establish APSR alignment.

A.2

Reduction of THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER ensures that local LHR increases, due to a misaligned APSR, will not cause the core design criteria to be exceeded. The required Completion Time of 2 hours allows the operator sufficient time for reducing THERMAL POWER.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification at a 12 hour Frequency that individual APSR positions are within 6.5% of the group average height limits allows the operator to detect an APSR beginning to deviate from its expected position. In addition, APSR position is continuously available to the operator in the control room so that during actual APSR motion, deviations can immediately be detected.

REFERENCES

1. SAR, Section 1.4, GDC 10 and GDC 28.
2. 10 CFR 50.46.
3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Position Indicator Channels

BASES

BACKGROUND

According to the SAR discussion of GDC 13 (Ref. 1), adequate instrumentation and controls are provided to maintain operating variables within prescribed ranges for normal operation and monitor accident conditions as appropriate to assure adequate safety. LCO 3.1.7 is required to ensure OPERABILITY of the CONTROL ROD and APSR position indicators, and thereby ensure compliance with the CONTROL ROD and APSR alignment and insertion limits.

The OPERABILITY, including position indication, of the CONTROL RODS is an initial condition assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment for the CONTROL RODS and APSRs is assumed in the safety analysis, which directly affect core power distributions and assumptions of available SDM.

Mechanical or electrical failures may cause a CONTROL ROD or APSR to become misaligned from its group. CONTROL ROD or APSR misalignment may cause increased local linear heat rates (LHRs), due to the asymmetric reactivity distribution, and a reduction in the total available CONTROL ROD worth for reactor shutdown. Therefore, CONTROL ROD and APSR alignment are related to core operation within design LHR limits and the core design requirement of a minimum SDM. CONTROL ROD and APSR position indication is needed to assess OPERABILITY and alignment.

Limits on CONTROL ROD and APSR alignment, and CONTROL ROD and APSR group position have been established, and all CONTROL ROD and APSR positions are monitored and controlled during operation to ensure that the power distribution and reactivity limits defined by the design LHR and SDM limits are preserved.

Three methods of CONTROL ROD and APSR position indication are provided in the Control Rod Drive Control System. The three means are by absolute position indicator, relative position indicator transducers, and zone reference indicators. The absolute position indicator transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the control rod drive mechanism (CRDM) motor tube extension. Switch contacts close when a permanent magnet mounted on the upper end of the CONTROL ROD or APSR assembly leadscrew extension comes near. As the leadscrew and CONTROL ROD or APSR move, the switches operate sequentially, producing an analog voltage proportional to position. Other reed switches included in the same tube with the absolute position indicator matrix provide full in and full out limit indications, and position indications at 0%, 25%, 50%, 75%, and 100% travel. This series of seven indicators are called zone reference indicators. The relative position indicator transducer is a potentiometer, driven by a step motor that produces a signal proportional to CONTROL ROD or APSR position, based on the electrical pulse steps that drive the CRDM.

CONTROL ROD and APSR position indicating readout devices located in the control room consist of single rod position meters on a position indication panel and group average position meters. A selector switch permits either relative or absolute position indication to be displayed on all of the individual position indication meters. Indicator lights are provided on the individual position indication panel to indicate when each CONTROL ROD or APSR is fully withdrawn, fully inserted, enabled, or transferred, and whether a rod position asymmetry alarm condition is present. Additional indicators show full insertion, full withdrawal, and enabled for motion for each CONTROL ROD and APSR group. The consequence of continued operation with an inoperable absolute position indicator or relative position indicator channel is a decreased reliability in determining CONTROL ROD and APSR position. Therefore, the potential for operation in violation of design LHR or SDM limits is increased.

APPLICABLE SAFETY ANALYSES

CONTROL ROD and APSR position accuracy is essential during power operation. LHR, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2) with CONTROL RODS or APSRs operating outside their limits undetected. CONTROL ROD and APSR positions must be known in order to verify the core is operating within the group sequence, overlap, design LHRs, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Safety Rod Insertion Limits"; LCO 3.2.1, "Regulating Rod Insertion Limits"; and LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"). The CONTROL ROD and APSR positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "CONTROL ROD Group Alignment Limits," and LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits"). CONTROL ROD and APSR positions are continuously monitored to provide operators with information that ensures the unit is operating within the bounds of the accident analysis assumptions.

In MODES 1 and 2 while critical, the CONTROL ROD and APSR position indicator channels satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3). In MODE 2 while subcritical, the CONTROL ROD and APSR position indicator channels satisfy Criterion 4 of 10 CFR 50.36.

LCO

LCO 3.1.7 specifies that one position indicator channel be OPERABLE for each CONTROL ROD and APSR.

This requirement ensures that CONTROL ROD and APSR position indication during MODES 1 and 2 and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channel ensures that inoperable, misaligned, or mispositioned CONTROL RODS or APSRs can be detected. Therefore, LHR and SDM can be controlled within acceptable limits.

APPLICABILITY

In MODES 1 and 2, OPERABILITY of the position indicator channel is required, since the reactor is, or is capable of, generating THERMAL POWER in these MODES. In MODES 3, 4, 5, and 6, Applicability is not required because the reactor is shut down with the required minimum SDM and is not generating significant THERMAL POWER.

ACTIONS

A.1

If the required position indicator channel is inoperable for one or more rods, the position of the CONTROL ROD or APSR is not known with certainty. Therefore, each affected CONTROL ROD or APSR must be declared inoperable, and the limits of LCO 3.1.4 or LCO 3.1.6 apply. The required Completion Time for declaring the rod(s) inoperable is immediately. Therefore LCO 3.1.4 or LCO 3.1.6 is entered immediately, and the required Completion Times for the appropriate Required Actions in those LCOs apply without delay.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

A CHANNEL CHECK of the required position indication channel ensures that position indication for each CONTROL ROD and APSR remains OPERABLE and accurate. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. However, this CHANNEL CHECK will be used to detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

When compared to other channels, the agreement criteria between channels is determined by the unit staff. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The CHANNEL CHECK supplements less formal but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required position indicator channel.

The required Frequency of 12 hours is adequate for verifying that no degradation in system OPERABILITY has occurred.

SR 3.1.7.2

A CHANNEL CALIBRATION of the required position indication channel verifies that the channel responds within the necessary range and accuracy.

The Frequency of 18 months is based on operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. SAR, Section 1.4, GDC 13.
2. SAR, Chapter 14.
3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions Systems - MODE 1

BASES

BACKGROUND

The purpose of this LCO is to permit PHYSICS TESTS to be conducted by providing exemptions from the requirements of other LCOs. Establishment of a test program to verify that structures, systems, and components will perform satisfactorily in service is required by Section XI of 10 CFR 50, Appendix B (Ref. 1). Testing is required as an integral part of the design, fabrication, construction, and operation of the unit. All functions necessary to ensure that specified design conditions are not violated during normal operation and abnormalities must be tested. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to:

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality; during startup, low power operations, and power ascension; at high powers; and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. 3).

The inclusion of this PHYSICS TESTS Exception LCO is acceptable based on the use of approved written procedures, administrative controls, the requirements of 10 CFR 50.59, and the LCO 3.1.8 provisions in effect during the conduct of PHYSICS TESTS. PHYSICS TESTS procedures are written and approved in accordance with established guidelines. The procedures include all information necessary to permit a detailed execution of testing required to ensure the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation. Examples of PHYSICS TESTS include determination of critical boron concentration, CONTROL ROD group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because reactor protection criteria are preserved by the LCOs still in effect and by the SRs. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on linear heat rate (LHR), ejected rod worth, and shutdown capability are maintained during the PHYSICS TESTS.

Reference 4 describes the initial testing of the facility, including PHYSICS TESTS. Table 13-2 (Ref. 5) summarizes the post-criticality tests. Requirements for reload fuel cycle PHYSICS TESTS are given in SAR Section 3A.9 (Ref. 3). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, one or more LCOs must sometimes be suspended to make completion of PHYSICS TESTS possible or practical.

This is acceptable as long as the fuel design criteria are not violated. When one or more of the limits specified in:

LCO 3.1.4, "CONTROL ROD Group Alignment Limits";
LCO 3.1.5, "Safety Rod Insertion Limits";
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits";
LCO 3.2.1, "Regulating Rod Insertion Limits";
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," for the restricted operation region only;
LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits"; and
LCO 3.2.4, "QUADRANT POWER TILT (QPT)"

are suspended for PHYSICS TESTS, the fuel design criteria are preserved by maintaining the LHR (in MODE 1 PHYSICS TESTS) within limits, maintaining ejected rod worth within limits by restricting regulating rod insertion to within the acceptable operating region or the restricted operating region, by limiting maximum THERMAL POWER and by maintaining SDM within the limit provided in the COLR. Therefore, surveillance of the LHR and SDM is required to verify that their limits are not exceeded. The limits for the LHR are specified in the COLR. Refer to the Bases for LCO 3.2.5 for a complete discussion of LHR. During PHYSICS TESTS, one or more of the LCOs that normally preserve the LHR limits may be suspended. However, the results of the safety analysis are not adversely impacted if verification that core LHRs are within their limits is obtained, while one or more of the LCOs is suspended. Therefore, SRs are placed on LHR during MODE 1 PHYSICS TESTS when THERMAL POWER exceeds 20% RTP to verify that the core LHRs remain within their limits. Periodic verification of these factors allows PHYSICS TESTS to be conducted while continuing to maintain the design criteria.

PHYSICS TESTS include measurement of core nuclear parameters or exercise of control components that affect process variables. Among the process variables involved are AXIAL POWER IMBALANCE and QPT, which represent initial condition input (power peaking) for the accident analysis. Also involved are the movable control components, i.e., the regulating rods and the APSRs, which affect power peaking. The limits for these variables are specified for each fuel cycle in the COLR.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36 (Ref. 6) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion for the other LCOs is provided in their respective Bases.

LCO

This LCO permits individual CONTROL RODS and APSRs to be positioned outside of their specified group alignment and withdrawal limits and to be assigned to other than specified CONTROL ROD groups, and permits AXIAL POWER IMBALANCE and QPT limits to be exceeded during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics and nuclear instrumentation operation.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1 (for the restricted operation region only, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. THERMAL POWER is maintained ≤ 85% RTP;

b. Nuclear overpower trip setpoint is ≤ 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP;

c. LHR is maintained within limits specified in the COLR while operating at greater than 20% RTP; and

d. SDM is verified to be within the limit provided in the COLR.

Operation with THERMAL POWER ≤ 85% RTP during PHYSICS TESTS provides an acceptable thermal margin when one or more of the applicable LCOs is out of specification. Eighty-five percent RTP is consistent with the maximum power level for conducting the intermediate core power distribution test specified in Reference 3. The nuclear overpower trip setpoint is reduced so that a similar margin exists between the steady state condition and trip setpoint as exists during normal operation at RTP.

LCO provision c is modified by a Note that requires the adherence to LHR requirements only when THERMAL POWER is greater than 20% RTP. This establishes an LCO provision that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

APPLICABILITY

This LCO is applicable in MODE 1, when the reactor has completed low power testing and is in power ascension, or during power operation with THERMAL POWER > 5% RTP but ≤ 85% RTP. This LCO is applicable for power ascension testing, as described in SAR Section 3A.9 (Ref. 3). In MODE 2, Applicability of this LCO is not required because LCO 3.1.9, "PHYSICS TESTS Exceptions - MODE 2," addresses PHYSICS TESTS exceptions initiated in MODE 2. In MODES 3, 4, 5, and 6, Applicability is not required because PHYSICS TESTS are not performed in these MODES.

ACTIONS

A.1 and A.2

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the unit conditions. Boration will be continued until SDM is within limit. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification. A Completion Time of one hour is provided for the operator to restore compliance with the excepted LCOs.

B.1

If THERMAL POWER exceeds 85% RTP, then 1 hour is allowed for the operator to reduce THERMAL POWER to within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCO, addressed by PHYSICS TESTS exceptions.

If the nuclear overpower trip setpoint is not within the specified limits, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCO, addressed by these PHYSICS TESTS exceptions.

If the results of the incore flux map indicate that LHR has exceeded its limit, then PHYSICS TESTS are suspended. This action is required because of direct indication that the core LHR, which is a fundamental initial condition for the safety analysis, is excessive. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

This Condition is modified by a Note that requires performance of the Required Action only when THERMAL POWER is greater than 20% RTP. This establishes an ACTIONS entry Condition that is consistent with LCO provision c and the Applicability of LCO 3.2.5, "Power Peaking."

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that THERMAL POWER is ≤ 85% RTP ensures that the required additional thermal margin has been established prior to and during PHYSICS TESTS. The required Frequency of once per hour allows the operator adequate time to determine any degradation of the established thermal margin during PHYSICS TESTS.

SR 3.1.8.2

Verification that core LHRs are within their limits ensures that core LHR and departure from nucleate boiling ratio will remain within their limits, while one or more of the LCOs that normally control these design limits are out of specification. The required Frequency of 2 hours allows the operator adequate time for collecting a flux map and for performing the LHR verification, based on operating experience. If SR 3.2.5.1 is not met, PHYSICS TESTS are suspended and LCO 3.2.5 applies. This Frequency is more conservative than the Completion Time for restoration of the individual LCOs that preserve the LHR limits.

This SR is modified by a Note that requires performance only when THERMAL POWER is greater than 20% RTP. This establishes a performance requirement that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

SR 3.1.8.3

Verification that the nuclear overpower trip setpoint is within the limit specified for each PHYSICS TEST ensures that core protection at the reduced power level is established during the PHYSICS TESTS. Performing the verification once prior to the performance of PHYSICS TESTS allows the operator adequate time for verifying the established trip setpoint before initiating PHYSICS TESTS.

SR 3.1.8.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. Reactor Coolant System (RCS) boron concentration;

b. CONTROL ROD position;

c. Doppler defect;

d. Fuel burnup based on gross thermal energy generation;

e. Samarium concentration;

f. Xenon concentration; and

g. Moderator defect.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. SAR, Section 3A.9.
4. SAR, Section 13.3, 13.4 and 13.6.
5. SAR, Section 13.4, Table 13-2.
6. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.9 PHYSICS TESTS Exceptions - MODE 2

BASES

BACKGROUND

The purpose of this MODE 2 LCO is to permit PHYSICS TESTS to be conducted by providing exemptions from the requirements of other LCOs. Establishment of a test program to verify that structures, systems, and components will perform satisfactorily in service is required by Section XI of 10 CFR 50, Appendix B (Ref. 1). Testing is required as an integral part of the design, fabrication, construction, and operation of the unit. All functions necessary to ensure that specified design conditions are not violated during normal operation and abnormalities must be tested. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to:

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality; during startup, low power operations, and power ascension; at high powers; and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. 3).

The inclusion of this PHYSICS TESTS Exception LCO is acceptable based on the use of approved written procedures, administrative controls, the requirements of 10 CFR 50.59, and the LCO 3.1.9 provisions in effect during the conduct of PHYSICS TESTS. PHYSICS TESTS procedures are written and approved in accordance with established guidelines. The procedures include all information necessary to permit a detailed execution of testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation.

Examples of MODE 2 PHYSICS TESTS include determination of critical boron concentration, CONTROL ROD group worth, and reactivity coefficients.

APPLICABLE SAFETY ANALYSES

Reference 4 describes the initial testing of the facility, including PHYSICS TESTS. Table 13-2 (Ref. 5) summarizes the post-criticality tests. Requirements for reload fuel cycle PHYSICS TESTS are given in SAR Section 3A.9 (Ref. 3). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more of the LCOs must be suspended to make completion of PHYSICS TESTS possible or practical.

It is acceptable to suspend the following LCOs for PHYSICS TESTS because reactor protection criteria are preserved by the LCOs still maintained and by the SRs:

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.4, "CONTROL ROD Group Alignment Limits";
LCO 3.1.5, "Safety Rod Insertion Limits";
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits";
LCO 3.2.1, "Regulating Rod Insertion Limits";
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality."

Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on THERMAL POWER and shutdown capability are maintained during the PHYSICS TESTS.

Shutdown capability is preserved by limiting THERMAL POWER and maintaining adequate SDM, when in MODE 2 PHYSICS TESTS. In MODE 2, the Reactor Coolant System (RCS) temperature must be within the narrow range instrumentation for unit control. The narrow range temperature instrumentation goes on scale at 520°F. Therefore, it is considered safe to allow the minimum RCS temperature to decrease to 520°F during MODE 2 PHYSICS TESTS, based on the low probability of an accident occurring and on prior operating experience.

PHYSICS TESTS include measurement of core nuclear parameters or exercise of control components that affect process variables.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36 (Ref. 6) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria for the other LCOs is provided in their respective Bases.

LCO

This LCO permits individual CONTROL RODS and APSRs to be positioned outside of their specified group alignment and withdrawal limits and to be assigned to other than specified CONTROL ROD groups during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics.

This LCO also allows suspension of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, and LCO 3.4.2, provided:

a. THERMAL POWER is ≤ 5% RTP;

b. Nuclear overpower trip setpoints on the OPERABLE nuclear power range channels are set to ≤ 5% RTP;

c. Nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit is OPERABLE; and

d. SDM is within the limit provided in the COLR.

The limits of LCO 3.2.3 and LCO 3.2.4 are not exempted by this specification because they do not apply in MODE 2. Inhibiting CONTROL ROD withdrawal, based on startup rate, also limits local linear heat rate (LHR), departure from nucleate boiling ratio (DNBR), and peak RCS pressure during accidents initiated from low power.

APPLICABILITY

This LCO is applicable when the reactor is either subcritical or critical with THERMAL POWER ≤ 5% RTP. The Applicability is stated as "during PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions. This LCO is applicable for initial criticality or low power testing, as described in SAR Section 3A.9 (Ref. 3). In MODE 1, Applicability of this LCO is not required because LCO 3.1.8, "PHYSICS TESTS Exceptions," addresses PHYSICS TESTS exceptions in MODE 1. In MODES 3, 4, 5, and 6, a test exception LCO is not required because the excepted LCOs do not apply in these MODES.

ACTIONS

A.1

If THERMAL POWER exceeds 5% RTP, a positive reactivity addition could be occurring, and a nuclear excursion could result. To ensure that local LHR, DNBR, and RCS pressure limits are not violated, the reactor is immediately tripped. The necessary prompt action requires manual operator action to open the control rod drive trip breakers without attempts to reduce THERMAL POWER by actuating the control system (i.e., CONTROL ROD insertion or RCS boration).

B.1 and B.2

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the unit conditions. Boration will be continued until SDM is within limit. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification. A Completion Time of one hour is provided for the operator to restore compliance with the excepted LCOs.

C.1

If the nuclear overpower trip setpoint is > 5% RTP, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification, in order to ensure that continuity of reactor operation is within initial condition limits. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

If the nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit function is inoperable, then 1 hour is allowed for the operator to restore the functions to OPERABLE status or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

The nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit function is not required when the reactor power level is above the operating range of the instrumentation channel. For example, if the reactor power level is above the source range channel operating range, then only the intermediate range high startup rate CONTROL ROD withdrawal inhibit is required to be functional.

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification that THERMAL POWER is ≤ 5% RTP ensures that local LHR, DNBR, and RCS pressure limits are not violated and that entry into Actions Condition A is performed promptly. Hourly verification is adequate for the operator to determine any change in core conditions, such as xenon redistribution occurring after a THERMAL POWER reduction, that could cause THERMAL POWER to exceed the specified limit.

SR 3.1.9.2

Verification that the nuclear overpower trip setpoint is within the limit specified for PHYSICS TESTS ensures that core protection at the reduced power level is established during PHYSICS TESTS. Performing the verification prior to the performance of PHYSICS TESTS allows the operator adequate time for verifying the established trip setpoint before initiating PHYSICS TESTS.

SR 3.1.9.3

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. CONTROL ROD position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Samarium concentration;

f. Xenon concentration;

g. Isothermal temperature coefficient (ITC), when below the point of adding heat (POAH);

h. Moderator defect, when above the POAH; and

i. Doppler defect, when above the POAH.

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical or critical but below the POAH, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. SAR, Section 3A.9.
4. SAR, Section 13.3, 13.4 and 13.6.
5. SAR, Section 13.4, Table 13-2.
6. 10 CFR 50.36.

CTS DISCUSSION OF CHANGES

ITS Section 3.1: Reactivity Control Systems

ADMINISTRATIVE

A1 The designated change represents a non-technical, non-intent change to the Arkansas Nuclear One, Unit 1 Current Technical Specifications (CTS) made to make the ANO-1 Improved Technical Specifications (ITS) consistent with the Babcock and Wilcox (B&W) revised Standard Technical Specification (RSTS), NUREG-1430, Revision 1. This change does not alter the requirements of the CTS or RSTS. Examples of this type of change include: wording preference; convention adoption; editorial, numbering and formatting changes; and hierarchy structure.

A2 The CTS Bases will be administratively deleted in their entirety in favor of the NUREG-1430 Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 The CTS 4.7.1.2 defined rod misalignment as being a deviation from the group average position of more than nine (9) inches. For consistency with the plant instrumentation and NUREG-1430, 6.5% will be used to establish CONTROL ROD and APSR misalignment in the ITS. ITS Bases B 3.1.4 includes reference to the fact that 9 inches and 6.5% are considered equivalent. This is consistent with NUREG-1430.

A4 Not used.

A5 The second statement of CTS 3.5.2.5.1 provides an exception to the requirement that all safety rods be fully withdrawn as stated in CTS 3.1.3.5. This allowance relaxes the requirement to shutdown, per CTS 3.1.3.7, when a safety rod is not fully withdrawn, provided the rod is inoperable per CTS 3.5.2.2. Through the adoption of ITS 3.1.5 and its associated ACTIONS, this allowance for continued operation of the unit with an inoperable and not fully withdrawn safety rod will be maintained. Although it is represented in a significantly different format, the requirements of CTS 3.5.2.5.1 are maintained by the requirements of the ITS. Due to the continuation of essentially equivalent requirements, this change is administrative in nature. This change is consistent with NUREG-1430.

A6 The requirement that a CONTROL ROD which cannot be exercised be declared inoperable, which is presented in the first statement in CTS 4.7.1.3, is maintained in the ITS through the requirements of ITS SR 3.1.4.2, CONTROL ROD freedom of movement verification, and the application of ITS SR 3.0.1. Although no specific ITS item is cross-referenced to this CTS item, the requirement is embodied in the structure and requirements of ITS Specifications 3.1.4 and 3.1.5, and the application of SR 3.0.1. The lack of a direct cross-reference represents no actual change in requirements and is administrative in nature.

A7 CTS 3.1.3.1 establishes the minimum temperature for criticality of 525°F except during low power physics testing when the requirements of CTS 3.1.8.3 shall apply. CTS 3.1.3.2 and CTS 3.1.8.3 establish a minimum temperature for criticality in accordance with the criticality curves provided on CTS Figure 3.1.2-2. CTS 3.1.3.2 and CTS 3.1.8.3 implicitly duplicate the requirements of CTS 3.1.2, "Pressurization, Heatup and Cooldown Limitations," which has an implied Applicability of "at all times." Because of the duplicative nature of CTS 3.1.3.2 and CTS 3.1.8.3, they have been administratively deleted. This is acceptable because these minimum temperature requirements will exist in ITS LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits." ITS 3.4.3 will have Applicability "at all times" and is not excepted by the Physics Testing exceptions provided by LCO 3.1.8, "PHYSICS TEST Exceptions - MODE 1." and LCO 3.1.9, "PHYSICS TEST Exceptions - MODE 2." Therefore, this minimum temperature for criticality requirement will continue to exist in the ITS.

A8 The intent of CTS 3.1.8.1.A and 3.1.8.1.B is to ensure that, during Low Power Physics Testing, all Reactor Protection System (RPS) Setpoints are maintained per the requirements of the RPS setpoints section of CTS (Table 2.3-1) with the exception of the nuclear overpower trip setpoint which shall be less than 5 percent. The distinction of specifying the requirements separately below 1720 psig and above 1800 psig is made to ensure that the requirements are clearly applicable whether RPS is in Shutdown Bypass (<1720 psig), or out of Shutdown Bypass (>1800 psig). The requirement to maintain the nuclear overpower trip setpoint at less than 5 percent is specified only when above 1800 psig because the Shutdown Bypass nuclear overpower trip setpoint specified in CTS Table 2.3-1 is also 5%. The adoption of ITS 3.1.9 and its Applicability will maintain requirements consistent with those found in CTS 3.1.8.1.A and 3.1.8.1.B. Since ITS 3.1.9 does not suspend the requirements of ITS 3.3.1, "Reactor Protection System (RPS) Instrumentation," it is clear that all applicable RPS setpoint requirements of ITS Table 3.3.1-1 apply even during MODE 2 PHYSICS TESTING. Additionally, ITS 3.1.9 provides the requirement that the "Reactor trip setpoints on the OPERABLE nuclear overpower channels are set to ≤ 5% RTP." This maintains a reactor trip setpoint requirement consistent with CTS 3.1.8.1.B. Finally, by allowing RPS overpower trip setpoints no higher than 5% RTP, CTS requirements ensured that this testing was performed at less than 5% RTP. The specified setpoints maintain requirements consistent with ITS 3.1.9.a.

Because the adoption of ITS 3.1.9, in lieu of CTS 3.1.8.1.A and 3.1.8.1.B, though significantly different in format, maintains requirements consistent with CTS 3.1.8.1.A and 3.1.8.1.B, this change is administrative in nature. This change does not result in any new requirements nor does it result in the removal of any current requirements.

A9 CTS 3.5.2.3 established a requirement that "the worth of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5." CTS 3.1.3.5 established requirements for safety rod withdrawal and limitations on regulating rod group insertion as established by Specification 3.5.2.5. The CTS did not explicitly establish a required action to verify that the potential ejected rod worth of a misaligned rod is within the assumptions used in the rod ejection analyses. However, it is an implicit requirement that CTS 3.5.2.3 would apply to misaligned CONTROL RODS. Therefore, CTS 3.5.2.3 is considered to embody the requirements of NUREG-1430 Required Action A.2.4 (ITS Required Action A.2.2.2).

A10 CTS 3.1.3.5 requires that the safety rod groups be fully withdrawn prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality. NUREG-1430 and ITS LCO 3.1.5 require that each safety rod be fully withdrawn during MODES 1 and 2. The NUREG and ITS are predicated on an "individual" rod basis and not a group position basis. Although this translates into an identical requirement to have all safety rods fully withdrawn in MODES 1 and 2, there will be no safety rod group position requirements or actions in the ITS, only individual safety rod requirements and actions. This change in presentation of requirements is considered administrative in nature and does not change the actual requirement that all safety rods be fully withdrawn during MODES 1 and 2. This change is consistent with NUREG-1430.

The Applicability for CTS 3.1.3.5 is "prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality." This statement, as applied at ANO-1, requires compliance with regulating rod insertion limits while in Hot Standby and Startup reactor operating conditions (equivalent to ITS MODE 2). Although not explicitly applied to Power Operations (MODE 1), this Specification must be applied during these conditions to preserve the SHUTDOWN MARGIN requirements. Because the Applicability of ITS 3.1.5 maintains requirements consistent with the Applicability of CTS 3.1.3.5, as applied at ANO-1, this change is administrative in nature and neither adds any additional requirements nor removes any existing requirements.

A11 CTS 4.7.1.2 requires that if a CONTROL ROD is misaligned from its group average position by more than 9 inches (6.5%), it shall be declared inoperable and the limits of CTS 3.5.2.2 shall apply. CTS 3.5.2.2 includes some actions which are applicable to all inoperable CONTROL RODS and some actions which are specifically applicable only to CONTROL RODS which are inoperable due to misalignment. Although ITS 3.1.4 and 3.1.6 differentiate between inoperable and misaligned rods, these Specifications are written in such a way as to provide appropriate actions to compensate for either case. (The specific discussion of the differences between the actions of CTS 3.5.2.2 and ITS 3.1.4 and 3.1.6 are contained in separate DOCs.) Through the adoption of ITS 3.1.4 and 3.1.6, the intent of CTS 4.7.1.2 which is to ensure that the appropriate actions are taken in the event that a CONTROL ROD or APSR becomes misaligned from its group average position is maintained. No new requirements are added by this change and the only requirement removed is the requirement to declare the misaligned rod inoperable based only on misalignment. This difference is a result of the difference in philosophy of implementation between the CTS and ITS. Therefore, this change is considered administrative and represents no significant change to the requirements for operating with a misaligned rod.

A12 CTS markup was annotated to show adoption of ITS 3.1.7 Actions Note. This change is administrative in that the Note is required by the format and usage associated with the structure and presentation of the Actions in NUREG-1430.

CTS DISCUSSION OF CHANGES

TECHNICAL CHANGE – MORE RESTRICTIVE

M1 CTS Specification 4.9 currently provides for the evaluation of reactivity anomalies during operation of the unit. The CTS requires that the reactivity anomaly be evaluated "periodically" by comparison of the actual boron concentration to the predicted boron concentration. Additional discussion of the process of anomaly determination is provided in the Bases of CTS Specification 4.9. This periodic evaluation is presently administratively controlled with a frequency of approximately once per month. Adoption of the NUREG-1430 Specification 3.1.2 will require that the Frequency be performed in accordance with a more restrictive schedule than that presently identified in the CTS. Specifically, ITS SR 3.1.2.1 will have a Frequency of "prior to entering MODE 1 after each fuel loading" and "31 EFPD thereafter" following 60 EFPD of cycle operation as established in the Note. These SR Frequencies are acceptable because they explicitly establish the time frame for the performance of the SR and are in accordance with current administrative practices. This change is consistent with NUREG-1430.

M2 CTS 4.9 provides for the evaluation of reactivity anomalies during operation of the unit. The CTS action requires that the reactivity anomaly be evaluated to determine the cause. No other specific power reduction or operating restriction is applied. ANO will adopt the NUREG-1430 LCO 3.1.2 ACTIONS with a specified Completion Time of 7 days for Condition A. This Required Action is more restrictive than the requirements established within the CTS. This change is appropriate because the Required Actions preserve the assumptions used in the accident analyses through the implementation of appropriate operating restrictions. This change is consistent with NUREG-1430.

M3 Not used.

M4 CTS 3.1.7.1 establishes the limits on Moderator Temperature Coefficient (MTC). The CTS states that the limits are applicable when "the reactor is not shutdown." The interpretation of this statement represents a condition where the reactor would be made 1% ΔK/K subcritical which represents a condition consistent with the CTS definition for Hot Shutdown. The slightly more restrictive Applicability of MODES 1 and 2 in ITS LCO 3.1.3 will provide requirements on MTC that are consistent with other reactivity control parameters in the ITS. This change is classified as slightly more restrictive due to the slight calculational difference that exists between a reactor shutdown by 1% ΔK/K and a reactor that has K_{eff} of less than or equal to 0.99. This change is consistent with NUREG-1430.

M5 CTS 3.5.2, "Control Rod Group and Power Distribution Limits," has a defined Applicability of "during power operation." However, these CONTROL ROD OPERABILITY requirements are in practice applied during both CTS Power Operation and Hot Standby operating conditions. The CONTROL ROD OPERABILITY criteria defined by CTS 3.5.2 will correlate with requirements in ITS 3.1.4, 3.1.5, 3.1.6, 3.2.1 and 3.2.2. All of these ITS Specifications have an Applicability of MODES 1 and 2. By specifying Applicability in MODE 2, in addition to MODE 1, requirements will exist in the ITS where none were previously specified in the CTS. This Applicability represents more restrictive operating requirements than those specified in the CTS. This change is necessary to ensure that CONTROL ROD OPERABILITY exists in MODES that are consistent with the ITS SHUTDOWN MARGIN requirements preserved by the CONTROL ROD alignment and positioning. This change is consistent with NUREG-1430.

M6 The requirements of NUREG-1430 LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," will be adopted as presented in ITS. No explicit requirements for SDM, as defined in ITS Section 1.1, at other than power operation conditions, exist in the CTS. When the RCS temperature was below the minimum temperature for criticality given in CTS 3.1.3.1, CTS 3.1.3.3 required a degree of subcriticality, based on the reactivity effect of depressurization, be maintained. In addition, there are subcriticality requirements contained in the CTS Section 1.0 definitions of Hot Shutdown, Cold Shutdown, and Refueling Shutdown. Adoption of ITS 3.1.1 is more restrictive in that specific LCO requirements, Required Actions, and Surveillance Requirements are established which were not previously, explicitly required in the CTS. This change is necessary to ensure that controls and compensatory measures are in place during MODES 3, 4, and 5 that ensure the subcriticality of the unit is maintained. This change is consistent with NUREG-1430.

M7 CTS 3.5.2.2.1 states "Operation with more than one inoperable rod ... shall not be permitted." The lack of a specified action time implies that CTS 3.0.3 applies. CTS 3.0.3 requires the unit to be in Hot Shutdown (ITS MODE 3) in 13 hours. The equivalent action established in NUREG-1430, LCO 3.1.4 Required Action C.2 and LCO 3.1.5 Required Action B.2, requires the unit to be in MODE 3 within 6 hours. ANO-1 will adopt these more restrictive requirements in order to provide explicit Completion Times where none are currently expressed. This change is consistent with NUREG-1430.

M8 The CTS requirement for performance of CONTROL ROD drop time testing is, per CTS 4.7.1.1, "following each refueling outage prior to return to power" and in Table 4.1-2 Item 1, "Each Refueling Shutdown." The NUREG-1430 SR 3.1.4.3 Frequency of "Prior to reactor criticality after each removal of the reactor vessel head" will be adopted to provide a test Frequency consistent with activities that have the potential of affecting the rod drop time. This change in Frequency imposes the additional requirement of performing CONTROL ROD drop time testing following any removal of the reactor vessel head not just following a refueling shutdown or outage. It additionally requires completion of this testing prior to criticality rather than "prior to return to power." Adoption of the ITS SR 3.1.4.3 Frequency is appropriate because it correlates the SR Frequency to the activity that has the greatest probability of affecting the CONTROL ROD capability and characteristics. This change is consistent with NUREG-1430.

M9 CTS 3.5.2.2.5 correlates to ITS 3.1.4 Required Action A.2.2.1. CTS 3.5.2.2.5 requires a reduction in power while operating with a misaligned CONTROL ROD; however, there is no specified Completion Time. ITS 3.1.4 Required Action A.2.2.1 similarly requires a reduction in THERMAL POWER, while operating with a misaligned CONTROL ROD, and includes the added restriction of a 2 hour Completion Time. The adoption of the Completion Time ensures conservative actions are expeditiously initiated to minimize the potential effects of power redistribution and subsequent power peaking. This change is consistent with NUREG-1430.

M10 The first two sentences of CTS 3.5.2.2.2 and the first sentence of CTS 3.5.2.2.3 correlate to ITS 3.1.4 Required Actions A.1.1, A.1.2, C.1.1, and C.1.2 with the exception of the second specified Completion Time for Required Action A.1.1. Therefore, the second Completion Time for ITS 3.1.4 Required Action A.1.1 is shown as being adopted. This addition will impose more stringent requirements on unit operation by specifying that SDM be verified on a 12 hour Frequency after the initial verification. While this is not a departure from current operating practices, it is an additional requirement not given in the CTS. This periodic verification of SDM is appropriate because of the potential effects associated with power level changes, power redistribution, and transient fission product poisons. This change is consistent with NUREG-1430.

M11 ITS SR 3.1.4.1, SR 3.1.5.1 and SR 3.1.6.1 requirements to verify that CONTROL RODS and APSRs are within 6.5% of their group average and that safety rods are fully withdrawn, on a 12 hour Frequency, has been adopted. No specific requirement for this verification is expressed in CTS. Current operating practice is to perform these verifications in conjunction with and on the same frequency as the check of the Absolute and Relative Position Indication instrumentation. This change is consistent with NUREG-1430.

M12 CTS 3.1.7.3 currently requires the unit to be placed "in at least HOT STANDBY" (reactor critical below 2% power) if the Moderator Temperature Coefficient (MTC) is outside its limits. The adoption of ITS 3.1.3 ACTION A will require the unit to be placed in MODE 3 if MTC is outside its limits. This conservative action is consistent with other ITS reactivity control Specifications and removes the unit from the Applicability established for ITS 3.1.3. This change is consistent with NUREG-1430.

M13 ITS 3.1.7 Applicability has been adopted. No explicit Applicability exists for the equivalent requirements found in CTS 4.7.1.3. The addition of the ITS 3.1.7 MODE 1 and 2 Applicability has been made to provide requirements for verification of CONTROL ROD and APSR position indication that are consistent with ITS LCO 3.1.4, 3.1.5, 3.1.6, 3.2.1 and 3.2.2 requirements governing CONTROL ROD positioning. This change is consistent with NUREG-1430.

M14 The CTS markup reflects the adoption of NUREG-1430 LCO 3.1.8 PHYSICS TESTS Exceptions - MODE 1 as it is presented in the ITS. The CTS excepted certain individual specifications with a statement such as "except for physics testing." [This is one frequent usage of the exception and is not intended to represent every usage of the exception in the CTS.] No differentiation was made in the CTS of the applicability of these exceptions with respect to the unit's THERMAL POWER level. Further, only a minimal number of specific requirements were presented in the CTS during the conduct of PHYSICS TESTS and no required actions were presented. ITS 3.1.8 LCO, ACTIONS and SRs have been shown as adopted to provide this power level (or MODE) dependency. Although the PHYSICS TEST exceptions existed in the CTS, the power level dependency did not exist. Thus, the ITS will result in more restrictive requirements. This change is consistent with NUREG-1430.

Additionally, the ACTIONS and SRs of ITS 3.1.9 PHYSICS TEST Exceptions-MODE 2 have been adopted. These items function to verify that the LCO requirements are satisfied and provide necessary remedial actions should the requirements not be satisfied. Because the CTS did not impose specific restrictions, required actions or additional surveillance requirements comparable to those established in the ITS, this change is more restrictive. The adoption of the additional requirements, Required Actions and SRs is appropriate due to the nature of PHYSICS TESTS. This change is consistent with NUREG-1430.

M15 ITS 3.1.2 Required Action A.2 and Required Action B.1 will be adopted. The Frequency of ITS SR 3.1.2.1 and the Notes modifying this SR are also adopted. The adoption of these requirements, where none existed previously, represents more restrictive requirements on the unit. These Required Actions provide appropriate guidance for continued unit operation with a reactivity anomaly that exceeds its limit and conservative action to place the unit in MODE 2 should the Required Actions and associated Completion Times of Condition A not be met. The SR Notes are necessary to provide guidance for completion of the SR. The SR Frequency adopted is appropriate to determine the presence of a reactivity anomaly shortly after unit startup but prior to significant unit operation with the anomalous condition. The adoption of the SR Frequency is specifically more restrictive because it specifies the performance of the SR "once prior to entering MODE 1 after each fuel loading." This change is consistent with NUREG-1430.

M16 The 72 hour Completion Time for ITS 3.1.4 Required Action A.2.2.2 (NUREG-1430 3.1.4 Required Action A.2.4) is shown on the CTS markup as being adopted in the ITS. This is more restrictive because no Completion Time was explicitly established in the CTS for the completion of ejected rod worth verification as required by CTS 3.5.2.3. The adoption of the Completion Time is appropriate to ensure that the verification is promptly initiated; thus, allowing implementation of compensatory measures, if appropriate. This change is consistent with NUREG-1430.

M17 The "no flow" rod drop time testing acceptance criteria is shown as being administratively deleted in the CTS 4.7.1.1 markup. This acceptance criteria and the conditions of the testing have not been demonstrated as being acceptable for satisfying the rod drop time surveillances that preserve the accident analysis assumptions. This allowance and its test criteria are not currently utilized by ANO-1. In fact SAR Section 3.A, does not allow completion of startup testing and entrance into MODE 1 without performing the full flow test. The deletion of this allowance from the CTS results in the ITS possessing more restrictive requirements than those established by the CTS. NUREG-1430 does not establish a similar "no flow" testing methodology or acceptance criteria, thus, this deletion of material is consistent with NUREG-1430.

M18 The CTS 4.7.1.2 provision that allowed the CONTROL ROD with the greatest deviation from the group average position to be evaluated first for the purpose of determining compliance with CTS requirements has been shown as administratively deleted. This allowance is not contained within nor does it support the requirements of NUREG-1430 or the ITS; thus, the ITS will be more restrictive than the CTS in this regard. Multiple CONTROL RODS deviating from the group average position are dealt with simultaneously in the ITS. The deletion of this CTS allowance is acceptable because of the conservative nature of the ITS in addressing multiple CONTROL ROD deviations from their group average position. This change is consistent with NUREG-1430.

M19 The CTS was annotated to show the adoption of ITS 3.1.4 Required Action A.2.2.3 with its Note (NUREG-1430 3.1.4 Required Action A.2.5) which will require verification of acceptable core linear heat rates (LHRs) during operation at less than or equal to 60% of the ALLOWABLE THERMAL POWER with a misaligned CONTROL ROD. This Required Action has a 72 hour Completion Time which is acceptable because core LHRs are limited by the THERMAL POWER reduction (ITS 3.1.4 Required Action A.2.2.1). The Required Action is preceded by a Note that specifies the Required Action is only required to be performed when THERMAL POWER is greater than 20% RTP. This establishes a requirement for verification of core power distribution during unit operation consistent with the OPERABILITY of the incore detector system. This change is consistent with NUREG-1430.

M20 The methodology specified in CTS 3.5.2.2.2 for restoring SDM, if it is determined to be less than adequate, allows boration to be secured once the worth of the inoperable rod has been met or once the limits of CTS 3.5.2.5.3 are met (i.e., the regulating rod groups are withdrawn above the SDM insertion limit curve given in the COLR). The ITS requirement will be that SDM be calculated and verified to be within the limit specified in the COLR taking into consideration the reactivity worth of the inoperable CONTROL ROD. Therefore, when addressing a single inoperable CONTROL ROD, the ITS will not allow boration to be secured once the regulating groups have been positioned above the SDM limits established by the regulating rod insertion curves given in the COLR. [Note: this discussion does not impact other CTS and ITS Specifications that would require continued boration should the regulating groups be inserted beyond their SDM insertion limits.] Thus, the ITS will be more restrictive because it will exclude an option for compliance that is present in the CTS. The ITS method of SDM verification is consistent with current operating practices, though not specified by CTS. The adoption of the ITS requirements is appropriate because the regulating rod group insertion limits curve given in the COLR was not derived such that SDM was preserved with an additional inoperable rod, nor is it intended to address this condition. This change is consistent with NUREG-1430.

M21 CTS 3.5.2.2.3 requires the unit to be placed in Hot Standby (reactor critical and <2% power) if the required SHUTDOWN MARGIN (SDM) can not be verified or obtained within 1 hour. The CTS does not establish a specific completion time for this required action. The adoption of ITS 3.1.4 ACTION B will require the unit be placed in MODE 3 (i.e., $K_{eff} < 0.99$) within 6 hours if adequate SDM is not verified within one hour or if boration is not initiated to obtain SDM within one hour. Thus, ITS 3.1.4 ACTION B is more restrictive than the corresponding CTS requirement in that it requires the unit be taken to a lower MODE as a result of failure to satisfy SDM requirements. These additional requirements are necessary to remove the unit from an operating condition when boration has been inadequate to restore the necessary SDM. This change is consistent with NUREG-1430.

M22 CTS 3.5.2.2.3 correlates to ITS 3.1.4 Required Action B.1. CTS 3.5.2.2.3 requires that the unit be placed in Hot Standby if the preceding CTS actions have been unsuccessful in restoring the required SDM. The CTS does not specify a Completion Time. ITS 3.1.4 Required Action B.1 similarly addresses the Required Actions should the preceding ITS actions not be successfully implemented, and includes the added restriction of a 6 hour Completion Time. The adoption of the Completion Time ensures conservative actions are initiated to remove the unit from the LCO Applicability. This change is consistent with NUREG-1430.

M23 CTS 3.5.2.2.6 correlates to ITS 3.1.4 Required Action A.2.1. These Specifications allow the unit to continue to operate at unrestricted power levels above 60% ATP provided the inoperable regulating rod can be positioned such that it is contained within the allowable group alignment limits and the associated group positioned within the allowed group insertion limits. The CTS does not specify a Completion Time for this action. However, ITS 3.1.4 Required Action A.2.1 includes the added restriction of a 2 hour Completion Time. The adoption of the Completion Time ensures conservative actions are initiated to minimize the potential affects of power redistribution and subsequent power peaking. This change is consistent with NUREG-1430.

M24 CTS 3.5.2.2.6 correlates to ITS 3.1.6 Required Action A. 1. These Specifications allow the unit to continue to operate at unrestricted power levels above 60% ATP provided the inoperable APSR can be positioned such that it is contained within the allowable group alignment limits. The CTS does not specify a Completion Time for this action. However, ITS 3.1.6 Required Action A.1 includes the added restriction of a 2 hour Completion Time. The adoption of the Completion Time ensures conservative actions are initiated to minimize the potential affects of power redistribution and subsequent power peaking. This change is consistent with NUREG-1430.

M25 CTS 3.5.2.2.6 specifies that operation above 60% of ALLOWABLE THERMAL POWER (ATP) may continue with an APSR inoperable due to misalignment (as established by CTS 4.7.1.2) if the group is positioned such that the rod is no longer misaligned. This action restores compliance with the LCO; thus, no further action is required and power operation is unrestricted. The CTS establishes no required action if the unit is below 60% ATP. Further, the CTS does not specifically state the required action should an APSR not be capable of being aligned within its group alignment limits. The ITS will require THERMAL POWER to be reduced to ≤ 60% of the ALLOWABLE THERMAL POWER with a Completion Time of 2 hours. This change will incorporate an action that is implied by the current license basis.

TECHNICAL CHANGE – LESS RESTRICTIVE

L1 The ITS SR 3.1.4.2 required Frequency is less restrictive than the CTS.
CTS Table 4.1-2 Item 2 requires movement of CONTROL RODS on a frequency of every two (2) weeks. The ITS Frequency will be 92 days. Based on the historical operating reliability of the CONTROL RODS, this change in Frequency from 14 days to 92 days is not considered to represent a significant reduction in the ability to verify system reliability. This position is supported by Generic Letter 93-05, "Line-Item

Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation." The reduction in Frequency of CONTROL ROD freedom of movement verification lessens the overall number of CONTROL ROD drive system manipulations (power supply transfers, safety rod movement, etc.) and thereby tends to lessen the overall likelihood of dropped CONTROL RODS which can occur due to failures of portions of the control rod drive system. Though not easily quantifiable, the reduction in the overall likelihood of producing a dropped CONTROL ROD, specifically those caused by a system failure during testing, represents an overall increase in the safety of the unit. This change is consistent with NUREG-1430.

L2 ITS SR 3.1.4.3 will be adopted in place of CTS 4.7.1.1. The adoption of ITS SR 3.1.4.3, including its NOTE, provides ANO-1 with the additional flexibility of testing CONTROL ROD drop times with reactor coolant flow conditions other than full flow and no flow. By restricting operation of the unit to the reactor coolant pump combination used during rod drop testing, reactor coolant flow conditions, in the event of a reactor trip, are assured to be similar to those during CONTROL ROD drop time testing and thereby the testing is bounding. This change is consistent with NUREG-1430.

L3 Testing to insure freedom of movement of "Each Rod" is required above Cold Shutdown by CTS Table 4.1-2, Item 2. This testing is currently applied to both the CONTROL RODS and APSRs. Similar testing of the CONTROL RODS only, will be required by ITS SR 3.1.4.2 and will be applicable only in MODES 1 and 2. The adoption of the NUREG-1430 SR will result in less restrictive requirements. Specifically, the adoption of ITS SR 3.1.4.2 will remove the CTS requirement to perform freedom of movement testing on the APSRs. The purpose of this testing is to ensure that CONTROL RODS are not mechanically bound and will therefore insert upon a reactor trip. Because the APSRs, by design, do not insert upon a reactor trip, this testing is not required on the APSRs. Further, the APSRs are not credited as providing any of the required SDM on a reactor trip. This change is consistent with NUREG-1430.

L4 The CTS 3.5.2.2.2 and 3.5.2.2.4 requirements to exercise the remaining CONTROL RODS, in the event that a CONTROL ROD is declared inoperable, have been removed to improve the consistency between NUREG-1430 and ITS. The intent of these requirements was to provide for testing which could detect if additional CONTROL ROD(S) were immovable. Industry experience indicates that CONTROL ROD movement testing has in only a limited number of cases, led to the determination that a CONTROL ROD was mechanically immovable. This determination that a CONTROL ROD is mechanically immovable is instead much more likely to be made during initial CONTROL ROD withdrawal or during drop time testing. By design, electrical problems which prevent movement of CONTROL RODS, generally, do not prevent the insertion of CONTROL RODS in the event of a reactor trip. Additionally, industry experience indicates that this testing can and has resulted in reactor trips and dropped rods. The relatively low likelihood that this testing will actually reveal the inability of a CONTROL ROD to insert upon a reactor trip, coupled with the unnecessary challenges to safety systems caused by reactor trips or dropped rods which can occur as a result of this testing supports its removal from CTS. This change is consistent with NUREG-1430.

Note: This change will not remove the requirement to perform routine freedom of movement verification of the CONTROL RODS on a Frequency of every 92 days in accordance with ITS SR 3.1.4.2.

L5 CTS 3.5.2.2.3 has been modified to be consistent with the requirements of ITS 3.1.4 Required Action B.1. CTS 3.5.2.2.3 requires the unit be placed in Hot Standby (i.e., reactor critical but THERMAL POWER < 2% RTP) if, after one hour, SDM had not been verified to be greater than or equal to that required by the COLR. This CTS action is required regardless of whether or not boration is in progress to establish the required SDM. ITS 3.1.4 allows continued operation after one hour, even if the required SDM has not been verified, provided boration to establish SDM has been initiated. The adoption of the ITS 3.1.4 requirements allow the unit staff to focus on the restoration of required SDM without the additional operator burden of performing a unit shutdown. The initiation of boration to establish SDM will, in most cases, result in a reduction in power level which requires significant attention from the operating staff. This reduction of power level, when further complicated by the existence of an inoperable or misaligned CONTROL ROD, significantly complicates the operation of the Control Rod Drive System. These complications require even more attention from the operating staff. In light of these complicating factors, the requirement to shutdown the unit within one hour while less than adequate SDM exists, provided boration has been initiated to establish SDM, is not in the best interest of safety; and therefore, is not being retained. This change is consistent with NUREG-1430.

L6 CTS 3.1.3.5 requires that all safety rod groups be fully withdrawn prior to and during the approach to criticality. CTS 3.1.3.7 provides the action requirements if CTS 3.1.3.5 is not met, unless otherwise excepted. CTS 3.1.3.7 requires the inserted safety rod group be withdrawn within 15 minutes or the reactor be placed in at least Hot Shutdown (MODE 3) within the next 15 minutes. These CTS actions are predicated on entire "group" being out-of-position while the unit is in its approach to criticality. Individual safety rod and multiple rod inoperability (due to misalignment, loss of position indication, or slow drop time) is addressed by the CTS 3.5.2 and CTS 4.7.1 series of Specifications.

NUREG-1430 and ITS LCO 3.1.5 require that each safety rod be fully withdrawn during MODES 1 and 2. The NUREG and ITS are predicated on an "individual" rod basis and not a group position basis. Although this translates into an identical requirement to have all safety rods fully withdrawn in MODES 1 and 2, there will be no safety rod group position requirements or actions in the ITS, only individual safety rod requirements and actions. Thus, the ITS will not include actions comparable to CTS 3.1.3.7 requirements. This results in the ITS providing less restrictive requirements than the CTS.

As an effort to highlight these changes, CTS 3.1.3.7 was marked to show ITS 3.1.5 Required Action A.2, which declares inoperable within 1 hour, a safety rod that is not fully withdrawn. This declaration results in the performance of ITS 3.1.4 Required Actions which also preserve shutdown margin while addressing the potential operational concerns associated with a misaligned rod.

The removal CTS 3.1.3.7 group action requirement is acceptable because the ITS will continue to provide safety rod positioning requirements consistent with accident analysis assumptions. Operation with multiple safety rods misaligned or not fully withdrawn will not be allowed in the ITS; just as it is not allowed in the CTS. ITS 3.1.5 Required Action B.2 will require unit to be placed in MODE 3 within 6 hours of entry into Condition B (more than one safety rod not fully withdrawn). This time is reasonable and is based on the time required for the operator to reduce THERMAL POWER from RTP to MODE 3 without challenging unit systems. It must be noted that the CTS 3.1.3.7 time frames to be in Hot Shutdown were based on the reactor being subcritical during the approach to criticality. This change is consistent with NUREG-1430.

L7 During Power Operation (MODE 1), CTS 3.5.2.1 provides the "available shutdown margin" requirement and the action requirements in the event that SHUTDOWN MARGIN (SDM) is not adequate. In the ITS, the combination of LCO 3.1.5, "Safety Rod Insertion Limits," LCO 3.2.1, "Regulating Rod Insertion Limits," and the individual CONTROL ROD OPERABILITY requirements of LCO 3.1.4, "CONTROL ROD Group Alignment Limits," preserve the SDM requirements while in MODES 1 and 2. Maintaining CONTROL RODS within these limits will provide assurance that sufficient negative reactivity is available for insertion upon a reactor trip. During unit operation with an inoperable CONTROL ROD, CTS 3.5.2.2.2 provides a requirement to verify adequate SDM and initiate boration if SDM requirements were not met. Similarly, in the ITS, LCO 3.1.4, "CONTROL ROD Group Alignment Limits," will provide Required Actions that preserve the SDM requirements. [The relationship of ITS 3.2.1, "Regulating Rod Insertion Limits," to CTS 3.5.2.1 will be discussed, as appropriate, as a part of the discussion of ITS 3.2.1.]

In the CTS, if the "available shutdown margin" is less than required, CTS 3.5.2.1 directs the operator to "immediately initiate and continue boration injection until the required shutdown margin is restored," and CTS 3.5.2.2.2 directs that an "evaluation shall be initiated immediately to verify the existence of an available shutdown margin greater than or equal to that specified in the COLR." In the ITS, if the LCO 3.1.4 and 3.1.5 requirements are not met, LCO 3.1.4 and 3.1.5 Required Actions A.1.1 and A.1.2; LCO 3.1.5 Required Actions B.1.1 and B.1.2; and LCO 3.1.4 Required Actions C.1.1 and C.1.2 require verification of adequate SDM and initiation of boration to restore adequate SDM within 1 hour of entry into the Condition. The adoption of ITS 3.1.4 and 3.1.5 Actions will represent a relaxation of the requirement to "immediately" initiate an action such as boration. This less restrictive requirement is acceptable because the 1 hour Completion Time is adequate for determining the SDM, and if necessary, allows the operator sufficient time to align the required valves and start the necessary pumps without unduly challenging the operator's ability to safely operate the unit. This change is consistent with NUREG-1430.

L8 CTS 3.5.2.2.3 requirements for determining SHUTDOWN MARGIN (SDM) have been modified by the adoption of the SDM definition in Section 1.1 of the ITS and its application in ITS 3.1.4 and 3.1.5. By CTS requirements, the reactivity worth of any inoperable rod, regardless of the reason for inoperability, has to be accounted for as if it will not insert into the core upon a reactor trip. The ITS will require that only the reactivity worth of CONTROL RODS which are not capable of being fully inserted into the core need be considered as penalties to SDM. The intent of the CTS requirement to consider the reactivity of an inoperable CONTROL ROD in the SDM calculation is to insure that the reactor is in fact subcritical, by the amount specified in the COLR, following the insertion of the CONTROL RODS upon a reactor trip. Provided the inoperability of a CONTROL ROD is not due to the fact that the rod is not capable of fully inserting into the core upon a reactor trip, the requirement to consider that rod incapable of inserting its negative reactivity upon a reactor trip is overly conservative. This change is consistent with NUREG-1430.

L9 The CTS markup was annotated to reflect that the Moderator Temperature Coefficient (MTC) requirements of ITS LCO 3.1.3 may be excepted during PHYSICS TESTS pursuant to the requirements of ITS LCO 3.1.9, "PHYSICS TESTS Exceptions - MODE 2." To satisfactorily determine the operational behavior and characteristics of the reactor following startup, it may be necessary to significantly increase RCS boron concentration to maintain required critical conditions. During the limited period of time that the elevated RCS boron concentrations may exist at higher than normal concentrations, the MTC may be more positive than that allowed by ITS LCO 3.1.3. It is acceptable to suspend the MTC LCO during PHYSICS TESTS in MODE 2 based on the usage of approved written procedures, administrative controls, the requirements of 10CFR50.59, and the ITS LCO 3.1.9 provisions in effect during the conduct of the PHYSICS TESTS. These exceptions accommodate LCO suspension to verify the fundamental characteristics of the nuclear reactor which is critical in demonstrating the adequacy of design, analytical models, and confirmation of analysis results. This change is consistent with NUREG-1430.

L10 The CTS markup was annotated to show the adoption of ITS LCO 3.1.8, "PHYSICS TEST Exceptions-MODE 1," and LCO 3.1.9, "PHYSICS TESTS Exceptions - MODE 2," allowances to suspend the requirements of ITS LCO 3.1.4, "CONTROL ROD Group Alignment Limits," and LCO 3.1.6, "APSR Alignment Limits," during the conduct of PHYSICS TESTS. These exceptions suspend certain ITS LCO requirements that did not have PHYSICS TESTS exceptions in the CTS. The adoption of these exceptions is acceptable based on approved written procedures, administrative controls, the requirements of 10CFR50.59, and ITS LCO 3.1.8 and LCO 3.1.9 provisions in effect during the conduct of PHYSICS TESTS. These exceptions accommodate LCO suspension to verify the fundamental characteristics of the nuclear reactor which is critical in demonstrating the adequacy of design, analytical models, and confirmation of analysis results. This change is consistent with NUREG-1430.

L11 CTS requirements for CONTROL ROD and APSR position indication instrumentation are presented in CTS 4.7.1.3 and in CTS Table 4.1-1, Items 23 and 24. CTS 4.7.1.3 requires that for a CONTROL ROD or APSR to be considered OPERABLE, it must be located with one of three specified channels of indication. CTS Table 4.1-1 Items 23 and 24 require shiftly (12 hour) channel checks of only two of the three channels of indication specified in CTS 4.7.1.3. Additionally, refueling frequency calibrations of only these two channels are required.

Adoption of ITS LCO 3.1.7 establishes a requirement that maintains the CTS requirement that each CONTROL ROD and APSR have one OPERABLE channel of position indication. Further, ITS SR 3.1.7.1 and SR 3.1.7.2, in lieu of CTS Table 4.1-1, Items 23 and 24, provide testing requirements that establish appropriate assurance that the instrumentation required by ITS LCO 3.1.7 is OPERABLE. The potentially confusing cross-channel comparison of the CHANNEL CHECK located in CTS 4.1-1 was removed to ensure that any one OPERABLE indication channel, which can be adequately surveilled, will satisfy the LCO. The removal of this CTS cross-channel comparison detail results in the ITS being less restrictive. This is acceptable because the requirement to perform a CHANNEL CHECK of the instrumentation used to satisfy the LCO requirement is present in the ITS as SR 3.1.7.1.

L12 Testing to insure freedom of movement of "Each Rod" is required above Cold Shutdown by CTS Table 4.1-2, Item 2. Similar testing of the CONTROL RODS will be required by ITS SR 3.1.4.2 and will be applicable only in MODES 1 and 2. The adoption of the NUREG-1430 SR will result in less restrictive requirements. Specifically, the adoption of ITS SR 3.1.4.2 will remove the CTS requirement to perform this testing on CONTROL RODS while in MODES 3 and 4. This change actually only removes the requirement to test the CONTROL RODS while in operational MODES in which OPERABILITY of the CONTROL RODS is <u>not</u> required. This change provides for the application of Surveillance Requirements consistent with the MODES of Applicability for the tested components and is consistent with NUREG-1430.

L13 Not used.

L14 The shutdown actions in CTS 3.1.9.3 are proposed for deletion. CTS 3.1.9.1 and CTS 3.1.9.2 establish limits for the concentration of dissolved gases in the reactor coolant. These dissolved gas limits are intended to prevent possible control rod drive and/or control rod damage during a trip by ensuring that the control rod drive pressure housing is filled with water. CTS 3.1.9.3 specified an action to check the vessel level instrument vent for the accumulation of undissolved gases should the limits be exceeded. This action would be performed with the reactor shutdown because of the vent's location on the reactor vessel head. These limits and this required verification will be relocated to the Technical Requirements Manual (TRM). Because the appropriate required action will be retained in the TRM, the additional CTS 3.1.9.3 actions to restore the dissolved gas concentration to within limits within 24 hours or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours are unnecessary in purpose because a shutdown is required to perform the action and unnecessarily restrictive in time frame. In addition, the damage mechanism to the control rod drive(s) or control rod(s) would not prevent the control rods from performing their intended design function during an abnormality or accident. The deletion of these actions is consistent with NUREG-1430 in that the NUREG established no requirements pertaining to dissolved gas concentrations in the reactor coolant.

CTS DISCUSSION OF CHANGES

LESS RESTRICTIVE – ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to the Bases, SAR, TRM or COLR. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications. The details of performance of the surveillances have generally been relocated to the TRM. Changes to the SAR, TRM, and COLR will be controlled by 10 CFR 50.59. This change is consistent with NUREG-1430.

CTS Location	New Location
3.1.7.2	Bases - SR 3.1.3.1
3.1.9.1	TRM
3.1.9.2	TRM
3.1.9.3	TRM
Figure 3.1.9-1	TRM
Table 4.1-3, Item 1.d	TRM
Table 4.1-3, Note 7	TRM
4.7.1.1	SAR - Section 7.2.2.2.1
4.7.1.2	Bases - B 3.1.4 LCO
4.7.1.3	Bases – B 3.1.7 Background

3.1.5
3.1.8
3.1.9

<LATER> (3.4A) 3.1.3 Minimum Conditions for Criticality — LATER

Specification

<LATER> (3.4A) 3.1.3.1 The reactor coolant temperature shall be above 525F except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply. — LATER (A7)

3.1.9 LCO

<LATER> (3.4A) 3.1.3.2 Reactor coolant temperature shall be to the right of the criticality limit of Figure 3.1.2-2. — LATER

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization. (M6)

<LATER> (3.4B) 3.1.3.4 The reactor shall be maintained subcritical by at least 1 percent Δk/k until a steam bubble is formed and an indicated water level between 45 and 305 inches is established in the pressurizer. — LATER

<LATER> (3.2)

3.1.8, 3.1.9 LCO — LATER

3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn and the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality. MODES 1 and 2 — (A10) — LATER — (A10)

3.1.5 LCO

<LATER> (3.2)

3.1.5 Appl

<LATER> (3.4B) 3.1.3.6 The reactor shall not be made critical until at least 2 of the 3 emergency-powered pressurizer heater groups are operable. With less than 2 of the 3 required heater groups operable, restore the required heater groups to operable status within 72 hours. If the required heater groups are not restored to operable status within 72 hours, be in hot shutdown within the following 12 hours. — LATER

3.1.5 RA A.2

<LATER> (3.4A+B) 3.1.3.7 With any of the above limits violated, restore the reactor to within the limit in 25 minutes or be in at least Hot Shutdown within the next 15 minutes. — L6 + LATER

safety rod not fully withdrawn; safety rod; 1 hour; or declare the safety rod inoperable

Bases

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. (1) Calculations show that above 525F the positive moderator coefficient is acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less than 525F is prohibited except where necessary for low power physics tests. — (A2)

The potential reactivity insertion due to the moderator pressure coefficient (2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent Δk/k.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated Δk/k would limit the magnitude of power excursion resulting from a reduction of moderator density.

3.1.1
3.1.5

A2

~~The requirement that the reactor is not to be made critical below the limits of Figure 3.1.2-2 provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.~~

~~If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.~~

~~The requirement for pressurizer bubble formation and specified water level when the reactor is less than one (1) percent subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a start-up accident and that the water level is above the minimum detectable level.~~

~~The requirement that 2 of the 3 emergency-powered pressurizer heaters be operable provides assurance that sufficient heater capacity (≥126 kw) is available to provide reactor coolant system pressure control during a loss of off-site power.~~

~~The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.~~

~~The requirement for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.~~

~~REFERENCES~~

~~(1) FSAR, Section 3~~

~~(2) FSAR, Section 3.2.2.1.5~~

3.1.1
3.1.3
3.1.9

MTC — A1

3.1.7 **Moderator Temperature Coefficient of Reactivity Specification**

3.1.3 LCO — 3.1.7.1 The moderator temperature coefficient (MTC) shall be non-positive whenever thermal power is ≥95% of rated thermal power and shall be less positive than 0.9 x 10^{-4} Δk/k/°F whenever thermal power is <95% of rated thermal power ~~and the reactor is not shutdown~~ MODES 1 and 2 — M4

3.1.3 Appl.

SR 3.1.3.1 3.1.7.2 The MTC shall be determined to be within its limits by in MODE 1 — A1 confirmatory measurements prior to initial operation ~~above 5% of rated thermal power~~ after each fuel loading. ~~MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the limits in 3.1.7.1 above.~~ — LA1 BASES

3.1.3 RA A.1 3.1.7.3 With the MTC outside any one of the above limits, be in at least HOT STANDBY within 6 hours. MODE 3 — M12

~~Bases~~

~~A non-positive moderator coefficient at power levels above 95% of rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power, the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of +0.9 x 10^{-4} Δk/k/°F corrected to 95% of rated power. The most limiting event for positive MTC, the Startup Accident, has been analyzed for a range of moderator temperature coefficients including +0.9 x 10^{-4} Δk/k/°F.~~ — A2

<Add 3.1.9 PHYSICS TESTS exception to LCO 3.1.3> — L9

<Add 3.1.1 SHUTDOWN MARGIN (SDM)> — M6

3.1.8 ~~Low Power~~ Physics ~~Testing Restrictions~~ Exceptions – MODE 2 A1

~~Specification~~

~~The following special limitations are placed on low power physics testing.~~ A1

3.1.8.1 Reactor Protective System Requirements

A8

3.1.9.a LCO — A. Below 1720 psig, shutdown bypass trip setting limits shall apply in accordance with Table 2.3-1.

3.1.9.b LCO — B. Above 1800 psig, nuclear overpower trip shall be set at less than 5.0 percent. Other settings shall be in accordance with Table 2.3-1.

3.1.9.c LCO — 3.1.8.2 Startup rate rod withdrawal hold (1) shall be in effect at all times.

3.1.8.3 ~~During low power physics testing the minimum reactor coolant temperature for criticality shall be to the right of the criticality limit of Figure 3.1.2-2.~~ A7

3.1.8.d LCO
3.1.9.d LCO — The shutdown margin shall be maintained greater than or equal to that specified in the COLR with the highest worth control rod fully withdrawn.

~~Bases~~

~~The above specification provides additional safety margins during low power physics testing.~~

~~REFERENCES~~

~~(1) FSAR, Section 7.2.2.1.3.~~ A2

<Add 3.1.8 LCO a, b, c with Note; Appl.; Actions SR 3.1.8.1; SR 3.1.8.2 with Note; SR 3.1.8.3; SR 3.1.8.4> M14

<Add 3.1.9 Appl> A8

<Add 3.1.9 ACTIONS and SRs> M14

<Add LCO 3.1.8 & LCO 3.1.9 PHYSICS TESTS exceptions to LCO 3.1.4 and LCO 3.1.6.> L10

3.1

TSI-313-07-00 A1

3.1.9 Control Rod Operation

Specification

3.1.9.1 The concentration of dissolved gases in the reactor coolant shall be limited to 100 std. cc/liter of water at the reactor vessel outlet temperature.

3.1.9.2 Allowable combinations of pressure and temperature for control rod operation shall be to the left of and above the limiting pressure versus temperature curve for a dissolved gas concentration of 100 std. cc/liter of water as shown in Figure 3.1.9-1.

3.1.9.3 In the event the limits of Specifications 3.1.9.1 or 3.1.9.2 are exceeded, the vessel level instrument vent shall be checked for accumulation of undissolved gases. The temperature, pressure, and dissolved gas concentration shall be restored to within their limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

LA1

TRM

L14

Bases

By maintaining the reactor coolant temperature and pressure as specified above, any dissolved gases in the reactor coolant system are maintained in solution.

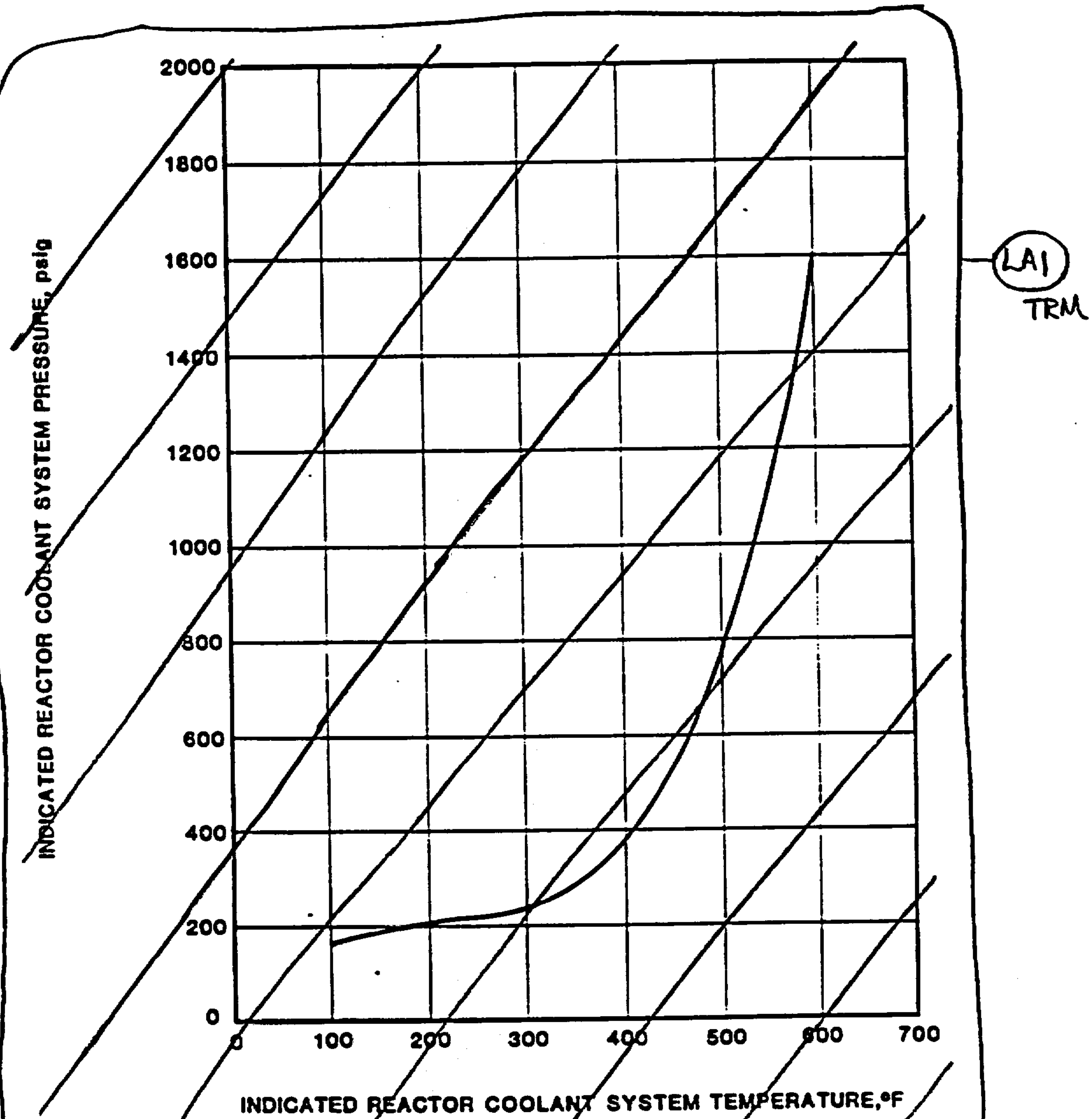
Although the dissolved gas concentration is expected to be approximately 20-40 std. cc/liter of water, the dissolved gas concentration is conservatively assumed to be 100 std. cc/liter of water at the reactor vessel outlet temperature.

The limiting pressure versus temperature curve for dissolved gases is determined by the equilibrium pressure versus temperature curve for the dissolved gas concentration of 100 std. cc/liter of water. The equilibrium total pressure is the sum of the partial pressure of the dissolved gases plus the partial pressure of water at a given temperature. The margin of error consists of the maximum pressure difference between the pressure sensing tap and lowest pressure point in the system, the maximum pressure gage error, and the pressure difference due to the maximum temperature gage error.

If either the maximum dissolved gas concentration (100 std. cc/liter of water) is exceeded or the operating pressure falls below the limiting pressure versus temperature curve, the vessel level instrument vent should be checked for accumulation of undissolved gases.

A2

3.1



Limiting Pressure VS Temperature for Control Rod Drive Operation With 100 STD CC/Liter H_2O

Figure 3.1.9-1

3.1.4
3.1.5
3.1.6

<Add 3.1.4 RA A.1.1 second Completion Time> M10

<Add 3.1.4 RA B.1 Completion Time> M22

3.5.2 Control Rod Group and Power Distribution Limits

3.1.4 Appl
3.1.5 Appl
3.1.6 Appl
& (LATER) (3.2)

Applicability

This specification applies to power distribution and operation of control rods during ~~power operation~~. MODES 1 and 2. M5 & LATER

Objective

~~To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.~~ A1

Specification

3.1.5 LCO — 3.5.2.1 The available shutdown margin shall be greater than or equal to that specified in the COLR with the highest worth control rod fully withdrawn. With the shutdown margin less than that required, ~~immediately~~ initiate and continue boration injection until the required shutdown margin ~~is restored~~. within 1 hr — & LATER, L7

3.1.5 RA A.1.1 & A.1.2
3.1.5 RA B.1.1 & B.1.2
& (LATER) (3.2)

3.5.2.2 Operation with inoperable rods:

3.1.4 R.A. C.2
3.1.5 R.A. B.2

1. Operation with more than one inoperable rod, as defined in Specification 4.7.1 and 4.7.2.3, in the safety or regulating rod groups shall not be permitted. Be in MODE 3 in 6 hours. M7

3.1.4 RA A.1.1
3.1.4 RA A.1.2
3.1.4 RA C.1.1
3.1.4 RA C.1.2

2. If a control rod in the regulating or safety rod groups is declared inoperable in the withdrawn position as defined in Specification 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existence of an available shutdown margin greater than or equal to that specified in the COLR. Boration may be initiated ~~either~~ to the worth of the inoperable rod ~~or until the regulating and transient rod groups are withdrawn to the limits of Specification 3.5.2.5.3, whichever occurs first.~~ ~~Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.~~ L7, M20, L4, L8

3.1.4 RA A.1.1, A.1.2, C.1.1, & C.1.2 Completion Time

any CONTROL ROD not capable of being fully inserted

3.1.4 RA B.1

and boration to restore SDM has not been initiated

3. If within one (1) hour of determination of an inoperable rod as defined in Specification 4.7.1, it is not determined that an available shutdown margin greater than or equal to that specified in the COLR exists combining the worth of ~~the inoperable rod~~ with each of the other rods, the reactor shall be brought to ~~the Hot Standby Condition~~ until this margin is established. MODE 3 in 6 hours M21, L5

4. ~~Following the determination of an inoperable rod as defined in Specification 4.7.1, all remaining rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.~~ L4

3.1.4 RA A.2.2.1 — 5. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, power shall be reduced to 60% of the ~~thermal power allowable for the reactor coolant pump combination~~. RTP A1

<Add 3.1.4 RA A.2.2.1 Completion Time> M9

<Add 3.1.4 RA A.2.2.3 with Note> M19

3.1.4
3.1.6
3.1.8

<Add 3.1.4 RA A.2.1 Completion Time> M23

3.1.4 RA A.2.1
3.1.6 RA A.1

6. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2 operation above 60 percent of the thermal power allowable for the reactor coolant pump combination may continue provided the rods in the group are positioned such that the rod that was declared inoperable is contained within allowable group average position limits of Specification 4.7.1.2 and the withdrawal limits of Specification 3.5.2.5.3.

3.1.4 RA A.2.2.2

3.5.2.3 The worth of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5. A9

LATER (3.2)

~~3.5.2.4 Quadrant Power Tilt:~~

3.1.8 LCO

LATER (3.2)

~~1. Except for physics tests, if quadrant power tilt exceeds the tilt limit set in the CORE OPERATING LIMITS REPORT, reduce power so as not to exceed the allowable power level for the existing reactor coolant pump combination less at least 2% for each 1% tilt in excess of the tilt limit.~~ LATER

3.1.8 LCO

LATER (3.2)

~~2. Within a period of 4 hours, the quadrant power tilt shall be reduced to less than the tilt limit except for physics tests, or the following adjustments in setpoints and limit shall be made:~~

~~a. The Protection System Maximum Allowable Setpoints for Axial Power Imbalance in the COLR shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.~~

~~b. The control rod group and APSR withdrawal limits shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.~~

~~c. The reactor power imbalance setpoints shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.~~ LATER

3.1.8 LCO

LATER (3.2)

~~3. If quadrant power tilt is in excess of 25%, except for physics tests or diagnostic testing, the reactor will be placed in the hot shutdown condition. Diagnostic testing during power operation with a quadrant power tilt is permitted provided the thermal power allowable for the reactor coolant pump combination is restricted as stated in 3.5.2.4.1 above.~~ LATER

~~4. Quadrant power tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15% of rated power.~~

<Add 3.1.4 RA A.2.2.2 Completion Time> M16
<Add 3.1.6 RA A.1 Completion Time> M24
<Add 3.1.6 RA A.2 Completion Time> M25

3.1.5
3.1.8
3.1.9

3.5.2.5 Control rod positions:

3.1.5 LCO NOTE

1. Technical Specification 3.1.3.5 (safety rod withdrawal) does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2. A5

<LATER> (3.2)

2. ~~Operating rod group overlap shall be 20% ±5 between two sequential groups,~~ except for physics tests. LATER

3.1.8 LCO
3.1.9 LCO

3.1.8
3.1.9

3.1.8 LCO
3.1.9 LCO

<LATER> (3.2)

3. Except for physics tests or exercising control rods, the control rod position setpoints are specified in the CORE OPERATING LIMITS REPORT for 4, 3, AND 2 pump operation. If the applicable control rod position setpoints are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within 4 hours. —LATER

3.1.8 LCO
3.1.9 LCO

<LATER> (3.2)

4. Except for physics tests or exercising axial power shaping rods (APSRs), the limits for APSR position are specified in the CORE OPERATING LIMITS REPORT.

With the APSRs outside the specified limit provided in the CORE OPERATING LIMITS REPORT, corrective measures shall be taken immediately to achieve the correct position. Acceptable APSR positions shall be attained within 4 hours. —LATER

<LATER> (3.2)

3.5.2.6 Reactor Power Imbalance: —LATER

1. Reactor power imbalance shall be monitored on a frequency not to exceed 2 hours during power operation above 40% rated power.

3.1.8 LCO

2. Except for physics tests, reactor power imbalance shall be maintained within the envelope defined by the CORE OPERATING LIMITS REPORT.

3. If the reactor power imbalance is not within the envelope defined by the CORE OPERATING LIMITS REPORT, corrective measures shall be taken to achieve an acceptable reactor power imbalance.

4. If an acceptable reactor power imbalance is not achieved within 4 hours, reactor power shall be reduced until reactor power imbalance setpoints are met.

<LATER> (3.2)

3.5.2.7 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the Superintendent. —LATER

Bases

The reactor power imbalance envelope defined in the CORE OPERATING LIMITS REPORT is based on either LOCA analyses (which have defined the maximum linear heat rate (see CORE OPERATING LIMITS REPORT), such that the maximum cladding temperature will not exceed the Final Acceptance Criteria) or loss of forced reactor coolant flow analysis (such that the hot fuel rod does not experience departure from nucleate boiling condition). Corrective measures will be taken immediately should the indicated quadrant power tilt, control rod position, or reactor power imbalance be outside their specified boundaries. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA or loss of forced reactor coolant flow occur is highly improbable because all of the power distribution parameters (quadrant power tilt, rod position, and reactor power imbalance) must be at their limits while —A2

3.1.8
3.1.9

simultaneously all other engineering and uncertainty factors are also at their limits.* Conservatism is introduced by application of:

a. Nuclear uncertainty factors.
b. Thermal calibration.
c. Fuel densification effects.
d. Hot rod manufacturing tolerance factors.
e. Fuel rod bowing.

The 20 ±5 percent overlap between successive control rod groups is allowed sinc the worth of a rod is lower at the upper and lower parts of the stroke. Contr rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Regulating
8	APSR (axial power shaping bank)

A2

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. As discussed above, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full-out position (1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than 0.65% Δk/k at rated power. These values have been shown to be safe by the safety analysis (2) of the hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% Δk/k is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% Δk/k at beginning of life, hot zero power, would result in a lower transient peak thermal power and therefore less severe environmental consequences than a 0.65% Δk/k ejected rod worth at rated power.

Control rod Groups are withdrawn in sequence beginning with Group 1. Groups 5, 6, and 7 are overlapped 20%. The normal position at power is for Groups 6 and 7 to be partially inserted.

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*Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument and calibration errors. The method used to define the operating limits is defined in plant operating procedures.

The quadrant power tilt limits set forth in the CORE OPERATING LIMITS REPORT have been established within the thermal analysis design base using the definition of quadrant power tilt given in Technical Specifications, Section 1.6. These limits in conjunction with the control rod position setpoints in the CORE OPERATING LIMITS REPORT, ensure that design peak heat rate criteria are not exceeded during normal operation when including the effects of potential fuel densification.

The quadrant power tilt limits and reactor power imbalance setpoints in the CORE OPERATING LIMITS REPORT, apply when using the plant computer to monitor the limits. The 2-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service. Additional uncertainty is applied to the limits when other monitoring methods are used.

During the physics testing program, the high flux trip setpoints are administratively set as follows to ensure that an additional safety margin is provided.

Test Power	Trip Setpoint %
0	<5
15	50
40	50
50	60
75	85
>75	105.5

A2

REFERENCES

(1) FSAR, Section 3.2.2.1.2

(2) FSAR, Section 14.2.2.2

Table 4.1-1 (Cont'd)

	Channel Description	Check	Test	Calibrate	Remarks	
<LATER> (3.3 B)	20. Reactor Building Spray System System Logic Channels	NA	M(1)	NA	(1) Including RB spray pump, spray valves, and chem. add. valve logic channels.	LATER
	21. Reactor Building Spray System Analog Channels					
	a. Reactor Building Pressure Channels	NA	M	R		LATER
<LATER> (3.3 D)	22. Pressurizer Temperature Channels	S	NA	R		
	23. Control Rod Absolute Position	S(1) SR 3.1.7.1	NA	R SR 3.1.7.2	(1) Compare with Relative Position Indicator	L1.1
	24. Control Rod Relative Position	S(1) SR 3.1.7.1	NA	R SR 3.1.7.2	(1) Check with Absolute Position Indicator	
<LATER> (3.5)	25. Core Flooding Tanks					LATER
	a. Pressure Channels	S	NA	R		
	b. Level Channels	S	NA	R		
<LATER> (3.3 D)	26. Pressurizer Level Channels	S	NA	R		LATER
	27. Makeup Tank Level Channels	D	NA	R		
<LATER> (3.3 D & 3.4 B)	28. Radiation Monitoring Systems other than containment high range monitors (item 57)				(1) Check functioning of self-checking feature on each detector.	LATER
	a. Process Monitoring System	S	Q	R		
<LATER> (3.3 D)	b. Area Monitoring System	S	M(1)	R		LATER
	c. Main Steam Line Radiation Monitors	S	M	R		

3.1.7

3.1.4
3.1.5
3.1.6

<Add SR 3.1.4.1> M1

<Add SR 3.1.5.1> M11

<Add SR 3.1.6.1> M11

Table 4.1-2
Minimum Equipment Test Frequency A1

	Item	Test	Frequency	
SR 3.1.4.3	1. Control Rods	Rod Drop Times of all Full Length Rods 1/	~~Each Refueling Shutdown~~ Following Reactor Vessel Head Removal	M8
SR 3.1.4.2	2. Control Rod Movement	Movement of Each ~~Rod~~ CONTROL ROD In MODES 1 and 2	Every ~~two weeks~~ 92 days ~~above Cold Shutdown Conditions~~	L1, L3, L12
<LATER> (3.4B)	3. Pressurizer Code Safety Valves	Setpoint	One Valve Every 18 Month	LATER
<LATER> (3.7)	4. Main Steam Safety Valves	Setpoint	Four Valves Every 18 Mon	LATER
	5. Refueling System Interlocks	Functioning	Start of Each Refueling Shutdown	R TRM
<LATER> (3.4B)	6a. Reactor Coolant System Leakage	Evaluate	Daily	LATER
	b. Reactor Coolant System Pressure Isolation Valves	Leakage Test Per Table 3.1.6.9	See Notes 1 & 2	
	7. Emergency-powered Pressurizer Heaters	Power availability	Daily	
		Heater capacity functional test	Every 18 Months	
<LATER> (3.6)	8. Reactor Building Isolation Trip	Functioning	Every 18 Months	LATER
<LATER> (3.7)	9. Service Water Systems	Functioning	Every 18 Months	LATER
	10. Spent Fuel Cooling System	Functioning	Every 18 Months when irradiated fuel is in the pool	R TRM

1/ Same as tests listed in Section 4.7 A1

<LATER> (3.4B)

Notes:

(1) Leak testing for each valve shall be individually accomplished to demonstrate operability following each refueling, following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement.

(2) Whenever integrity of a pressure isolation valve listed in Table 3.1.6.9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.

LATER

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency
1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)
	b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(7)
	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)
	d. Dissolved Gases	d. Weekly (7)
	e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)
	f. Boron Concentration	f. 3 times/week
	g. Radiochemical Analysis for Ē Determination (2) (4)	g. Monthly (7)
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup
3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)
5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10)
	b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)
6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup

⟨LATER⟩ (3.4B) ⟨LATER⟩ (3.9A) ⟨LATER⟩ (3.9) ⟨LATER⟩ (3.4B) ⟨LATER⟩ (3.5) ⟨LATER⟩ (3.7) ⟨LATER⟩ (3.6)

LATER LA1 TRM LATER LATER & R TRM LATER LATER LATER R TRM LATER LATER

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units µCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever the gross radioactivity concentration exceeds 10% of the limit specified in the Specification 3.1.4.1 or increases by 10 µCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

⟨LATER⟩ (3.4B) LATER

(2) ~~A radiochemical analysis shall consist of the quantitative measurement the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes shall be used in the determination of Ē. A radiochemical analysis and calculation of Ē and iodine isotopic activity shall be performed if the measured gross activity changes by more than μCi/gm from the previous measured level. The gamma energy per disintegration for those radioisotopes determined to be present shall be as given in "Table of Isotopes" (1967) and beta energy per disintegration shall be as given in USNRDL-TR-802 (Part II) or other references using the equivalent values for the radioisotopes.~~

(3) ~~In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 μCi/gm from the previous measured level.~~

<LATER> (3.4B) LATER

(4) ~~Iodine isotopic activities shall be weighted to give I-131 dose equivalent activity.~~

<LATER> (3.4B, 3.7) LATER

(5) ~~In addition to the weekly measurement, the radioiodine concentration shall be determined if there are indications that the primary to secondary coolant leakage rate has increased by a factor of 2.~~

R TRM

(6) ~~Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 1 percent but less that 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall taken within 24 hours of any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by the above.~~

~~Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken prior to any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by above.~~

<LATER> (3.4B) LATER

(7) ~~Not required when plant is in the cold shutdown condition or refueling shutdown condition.~~

<LATER> (3.4B, 3.7) & R TRM & LATER

(8) ~~O_2 analysis is not required when plant is in the cold shutdown condition or refueling shutdown condition.~~

<LATER> (3.4A) LA1 TRM LATER

(9) ~~Required only when fuel is in the pool and prior to transferring fuel the pool.~~

<LATER> (3.7) LATER

(10) ~~Not required when not generating steam in the steam generators.~~

<LATER> (3.7) R TRM & LATER

(11) ~~The following shall be required until the end of Cycle 2 operation:~~

a. ~~Gross radioiodine shall be determined at least three times per week during power operation.~~

<LATER> (3.4B) LATER

3.1.4
3.1.6
3.1.7

<Add SR 3.1.4.3 Note> L2

4.7 REACTOR CONTROL ROD SYSTEM TESTS

4.7.1 Control Rod Drive System Functional Tests

Applicability

Applies to the surveillance of the control rod system.

Objective

To assure operability of the control rod system.

Specification

A1

criticality L2

4.7.1.1 The control rod trip insertion time shall be measured for each control rod at either full flow or no flow conditions following each refueling outage prior to return to power. The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the Axial Power Shaping Rods (APSRs), from the fully withdrawn position to 3/4 insertion (104 inches travel) shall not exceed 1.66 seconds at reactor coolant full flow conditions or 1.20 seconds for no flow conditions. For the APSRs it shall be demonstrated that loss of power will not cause rod movement. If the trip insertion time above is not met, the rod shall be declared inoperable.

reactor vessel head removal

SR 3.1.4.3

M8 M17 LA1 SAR

6.5%

4.7.1.2 If a control rod is misaligned with its group average by more than an indicated nine (9) inches, the rod shall be declared inoperable and the limits of Specification 3.5.2.2 shall apply. The rod with the greatest misalignment shall be evaluated first. The position of a rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments.

3.1.4 LCO
3.1.6 LCO

A3 A11 M18 LA1 BASES

A6

4.7.1.3 If a control rod cannot be exercised, or if it cannot be located with absolute or relative position indications or in or out limit lights, the rod shall be declared to be inoperable.

3.1.7 RA A.1

LA1 BASES

Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has completed 104 inches of travel from the fully withdrawn position. The specified trip time is based upon the safety analysis in FSAR, Section 14, whose calculations are based on a rod drop from fully withdrawn to 2/3 inserted. Since the most accurate position indication is obtained from the zone reference switch at the 3/4 inserted position, this position is used instead of the 2/3 inserted position for data gathering.

Each control rod drive mechanism shall be exercised by a movement approximately two (2) inches of travel every two (2) weeks. This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanisms in this manner provides assurance of reliability of the mechanisms.

A rod is considered inoperable if it cannot be exercised, if the trip insertion time is greater than the specified allowable time, or if the rod

A2

<Add 3.1.7 LCO> L11
<Add 3.1.7 Appl.> M13
<Add 3.1.7 Actions Note> A12

3.1.4

deviates from its group average position by more than nine (9) inches. Conditions for operation with an inoperable rod are specified in Technical Specification 3.5.2.

REFERENCE

(1) FSAR, Section 14

A2

3.1.2 — 4.9 REACTIVITY ~~ANOMALIES~~ BALANCE A1

~~Applicability~~

~~Applies to potential reactivity anomalies.~~

~~Objective~~ A1

~~To require the evaluation of reactivity anomalies of a specified magnitude occurring during the operation of the unit.~~ A1

MODES 1 and 2

3.1.2 Appl — Specification

SR 3.1.2.1 — Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. M1

3.1.2 LCO — If the difference between the observed and predicted steady-state concentrations reaches the

3.1.2 RA A.1 — equivalent of one percent in reactivity, an evaluation of this abnormal occurrance will be made within 7 days to determine the cause of the discrepancy. M2

~~Bases~~

~~To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10 percent of the total core burnup. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1 percent $\Delta k/k$ would be unexpected, and its occurrence would be thoroughly investigated and evaluated.~~

~~The value of 1 percent $\Delta k/k$ is considered a safe limit since a shutdown margin of at least 1 percent $\Delta k/k$ with the most reactive rod in the fully withdrawn position is always maintained.~~ A2

<Add 3.1.2 RA A.2> M15

<Add 3.1.2 RA B.1> M15

<Add SR 3.1.2.1 NOTE and Frequency with NOTE> M15

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 3.1: Reactivity Control Systems

Entergy Operations has evaluated these proposed Technical Specification changes and has determined that they involve no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as indicated below:

<u>3.1 L1</u>

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

A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and the proposed Frequency has been determined to be adequate to demonstrate reliable operation of the equipment. This position is supported by Generic Letter 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation." Further, the Frequency of performance of a surveillance does not significantly increase the consequences of an accident because a change in Frequency does not change the assumed response of the equipment in performing its specified mitigation functions from that considered with the original Frequency. The control rods are used to support mitigation of the consequences of an accident; however, industry experience has shown a Frequency of 92 days is sufficient to detect failures in the Rod Control System and other information is available to the operator, e.g., individual rod position, to identify abnormalities. Therefore, this change does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for all equipment considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Changes in the surveilled parameter have been determined to be relatively slow during the proposed intervals, and the proposed Frequency has been determined to be sufficient to identify significant impact on compliance with the assumed conditions of the safety analysis. In addition, other indications continue to be available to indicate potential noncompliance. Therefore, an extended surveillance interval does not involve a significant reduction in the margin of safety.

3.1 L2

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

The control rods are used to support mitigation of the consequences of an accident; however, the reactor coolant system (RCS) flow conditions during control rod drop time verification are not considered an initiator of any previously analyzed accident. As such, the proposed change in the allowed RCS flow conditions will not significantly increase the probability of any accident previously evaluated. The proposed changes allow for testing the control rod drop times with less than a full complement of reactor coolant pumps operating. However, the operation of the plant is restricted to the pump combinations providing maximum flow less than or equal to the pump flow used for the testing. Therefore, the drop times verified during testing will remain valid for mitigating the consequences of any accident previously evaluated. Therefore, this change does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to ensure that the control rods are available for insertion of reactivity in the time frames consistent with the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety provided in the acceptable control rod drop times continues to be provided since these drop times have not been changed. The surveillance methodology is revised to allow testing with one, two, or three pumps operating. However, the operation of the plant is restricted to the reactor coolant pump combinations which maintain the margin of safety, i.e., those pump combinations providing maximum flow less than or equal to the pump flow used for the testing. Therefore, this change does not involve a significant reduction in a margin of safety.

3.1 L3

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

This change does not result in any hardware changes. Because the APSRs are not designed to insert on a reactor trip and are not credited toward the required shutdown margin, the removal of the requirement to verify Axial Power Shaping Rod (APSR) freedom of movement does not alter the functional performance characteristics of the control rods in performing their assumed safety analysis function.. As such the proposed change will not significantly increase the probability of any accident previously evaluated. Neither will the change alter the assumed function of the control rods in providing their assumed safety analysis function. Nor will this change alter the requirement to perform a freedom of movement verification of the control rods. Therefore, the proposed change does not involve a significant increase to 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still require proper demonstration of control rod OPERABILITY, consistent with applicable safety analysis assumptions and regulatory guidance. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change removes the required periodic verification that the APSR is moveable. The change does not alter the assumed function of the APSR or the operational restrictions and the administrative controls associated with the APSRs. Nor does the change alter the ability of the control rods to satisfy the safety analyses assumed function. As such, this change does not involve a significant reduction in a margin of safety.

3.1 L4

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

This change does not result in any hardware changes or changes in operating practice. The change removes an unnecessary additional performance of a surveillance which has been performed within its normally required Frequency. Not performing the surveillance would not affect any equipment which is assumed to be an initiator of any analyzed event. Further, since the surveillance continues to be performed on its normal Frequency there is no impact on the capability of the system to perform its required safety function. Therefore, the proposed change does not involve a significant increase in the probability or 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure adequate surveillance is performed to identify any degradation of the control rod freedom of movement. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The normal surveillance Frequency has been shown, based on operating experience, to be adequate for assuring the equipment is available and capable of performing its intended function. Additionally, the requirements of SR 3.0.4 (CTS 4.0.4) provide assurance the equipment is OPERABLE prior to beginning the functions for which it is required. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

<u>3.1 L5</u>

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

An immediate shutdown of the unit is not considered the most appropriate action for a loss of shutdown margin since such an action may result in diminished control capability. Rather, ACTIONS are proposed which will allow boration to restore the required shutdown margin to continue beyond the one hour time frame specified in the CTS. Neither an inoperable control rod nor inadequate shutdown margin have been considered as initiators for any accident previously evaluated. Therefore, an extended time frame in these conditions will not involve a significant increase in the probability of any previously evaluated accident. An extension of the Completion Time for the performance of the Required Action will not alter the capability of the mitigatory structures, systems or components from that assumed in establishing the Completion Time in the current Technical Specifications (CTS). Therefore, any consequences considered in the acceptance of the CTS Completion Time will not be significantly increased as a result of the adoption of the ITS Completion Time.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper ACTIONS are taken for an inoperable control rod resulting in a loss of shutdown margin. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

If an inoperable control rod results in a loss of shutdown margin, forcing a shutdown of the reactor may diminish the remaining control capability. However, allowing a short period to restore the required shutdown margin will, with high probability, result in restoration of the lost shutdown margin. This alternate action will also minimize the potential for plant transients that can occur during unit shutdown. As such, any perceived reduction in a margin to safety associated with the extended operating period will be offset with the benefit gained in avoiding a potential plant transient. Therefore, this change does not involve a significant reduction in a margin of safety.

3.1 L6

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

An extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time for performance of the Required Action also does not significantly increase the probability of occurrence of any analyzed event since the function of the equipment, or limit for the parameter, does not change. Further, the extension of the Completion Time is not associated with the assumed initiation of any evaluated event. Also, an extension of the Completion Time provides additional opportunity to restore compliance with the requirements and avoid the increased potential for a transient during the shutdown process. The Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because a change in the Completion Time does not change the assumed response of the equipment in performing its specified mitigation functions. Nor does the extension in the Completion Time significantly change the response of the core parameters to assumed scenarios, from that considered during the original Completion Time. Thus, the extension in Completion Time will not result in either a significant increase in probability or consequences of any evaluated accident.

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

The proposed extension of the Completion Time does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed extension in Completion Time has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, the extension of the Completion Time interval does not involve a significant reduction in the margin of safety.

3.1 L7

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

An extension of the Completion Time for Required Actions to verify adequate SHUTDOWN MARGIN (SDM), or completion of actions necessary to establish boration used to restore adequate SDM, are themselves not initiators of any evaluated accident. This change does not result in any hardware changes or physical alteration of the unit. Thus, the Completion Time for performance of the Required Action does not significantly increase the probability of occurrence of any analyzed event since the function of the limit on SDM does not change. The Completion Time for performance of Required Actions does not significantly increase the consequences considered while establishing the CTS Completion Time because the extension in the Completion Time does not change the assumed response of any structure, system or component in performing its specified mitigatory function from that considered during approval of the original CTS Completion Time.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, the extension of the Completion Time interval does not involve a significant reduction in the margin of safety.

3.1 L8

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

The CONTROL RODS are used to support mitigation of the consequences of an accident; however, the mitigation is supported by all CONTROL RODS which are capable of inserting into the core when required. A method of determining SHUTDOWN MARGIN which does not consider the availability of all such rods, except the assumed stuck rod, is overly conservative. Further, such inoperable CONTROL RODS are not considered an initiator of any previously analyzed accident. As such the proposed change in the method of determining the SHUTDOWN MARGIN with inoperable, but trippable, CONTROL RODS will not significantly increase the probability of any accident previously evaluated. The proposed change allows for consideration of all trippable CONTROL RODS, except one assumed stuck rod, in the determination of SHUTDOWN MARGIN. This is consistent with the analysis for determining the consequences of previously analyzed accidents. Therefore, this change does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to provide adequate SHUTDOWN MARGIN to assure the reactor is subcritical following a reactor trip. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety provided by the defined SHUTDOWN MARGIN continues to be provided consistent with the safety analyses when considering all trippable CONTROL RODS. Therefore, this change does not involve a significant reduction in a margin of safety.

3.1 L9

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

The parameter of moderator temperature coefficient (MTC) is an initial assumption of the safety analyses, but it is not considered as an initiator of any previously analyzed accidents. As such, the allowed increase in MTC during MODE 2 physics testing will not involve a significant increase in the probability of any previously evaluated accident. Because of the impact of MTC on reactivity control during an event, a change in MTC alone may significantly impact analyzed consequences of accidents. However, the preservation of SHUTDOWN MARGIN requirements, limitations on THERMAL POWER generation, and adherence to approved, written procedures whose requirements were evaluated under 10CFR50.59, compensate for the potential increase in MTC above its limits for the short duration of physics testing. Therefore, the allowed increase in MTC during MODE 2 physics testing will not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to preserve the reactor protection criteria. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The allowed increase in MTC during MODE 2 physics testing may result in a small reduction of the margin of safety for this specific parameter; however, the other parameters controlled by the physics testing exception LCO, along with the other unchanged LCO requirements, the preservation of SHUTDOWN MARGIN requirements, limitations on THERMAL POWER generation, and adherence to approved, written procedures whose requirements were evaluated under 10CFR50.59, are sufficient to prevent a significant decrease in a margin of safety.

3.1 L10

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

The requirement for individual control rod and axial power shaping rod alignment with their respective group average position is an initial assumption of the safety analyses. Further, individual control rod and axial power shaping rod misalignments are evaluated events in the accident analyses. However, the purpose of these physics testing exceptions is to specifically allow the measure and verification of fundamental core operating characteristics under careful, administratively controlled conditions, so as to confirm the adequacy of the design methods and models used to establish the operating limits for the unit. Because of the impact the control rod and axial power shaping rods potentially have on core reactivity conditions and core power distribution conditions, specific requirements on SDM and core THERMAL POWER levels are established. Specific LCO provisions and Required Actions have been established should the physics testing LCO provisions not be met. Thus, the allowed exceptions to the control rod and axial power shaping rod alignment limits during physics testing will not involve a significant increase in the probability of any accident previously evaluated. During the conduct of the physics testing with the control rod and axial power shaping rod alignment limits not met, adverse conditions may exist such that reactivity control and power distribution would adversely affect the consequences of certain postulated accidents. However, other parameters are additionally limited during the proposed physics testing and specific THERMAL POWER limitations are imposed to compensate for the potential increase adverse consequences. Therefore, the allowed exceptions to control rod and axial power shaping rod alignment requirements during physics testing will not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to preserve the reactor protection criteria. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The allowed exceptions to control rod and axial power shaping rod alignment limits during physics testing may result in a small reduction of the margin of safety for specific parameters; however, the other parameters controlled by the physics test exception LCO along with the other unchanged LCO requirements, are sufficient to preserve the available margins of safety before exceeding the reactor protection criteria. Therefore, this change does not involve a significant reduction in a margin of safety.

3.1 L11

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

The detail concerning performance of a CHANNEL CHECK on the required channel of position indication is not associated with the initiation of any evaluated accident. Thus, the removal of this detail will not alter the assumed frequency of initiation of an evaluated accident. In addition, the removal of this detail will not allow unit operation in a manner other than that presently allowed. Further, no reduction in requirements will exist with regard to the requirement to determine rod position. Thus, the removal of detail concerning the performance of a CHANNEL CHECK will not result in a significant increase in the probability of any evaluated accident. The detail associated with the performance of the CHANNEL CHECK does not serve a mitigatory function and does not alter the assumed ability to verify OPERABILITY of the required position indication channel. As long as the rod positions are determined to be within limits using OPERABLE position indication channels, the detail of the performance of the CHANNEL CHECK does not impact analyzed consequences of accidents. Therefore, the removal of the detail regarding performance of the CHANNEL CHECK does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to preserve the reactor protection criteria. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety for control rods is provided by the position of the rods, not the position indication. As long as the position of the rod can be accurately determined, the reactor protection criteria are preserved. The removal of the CHANNEL CHECK detail will not alter the ability to determine the rod position. Therefore, this change does not involve a significant reduction in a margin of safety.

3.1 L12

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

The removal of the requirement to perform a freedom of movement verification on CONTROL RODS while in MODES 3 and 4 does not result in any hardware changes, result in a physical alteration of the plant, or involve a change in the controls governing normal operation. The deletion of this requirement in MODES 3 and 4 removes a surveillance requirement applied to components that are not required to be OPERABLE in MODES 3 and 4. Thus, the removal of this requirement in these operational MODES does not alter the assumed initiation of any evaluated accident. Hence, this change does not involve a significant increase in probability. Further, the CONTROL ROD freedom of movement in MODES 3 and 4 is not associated with the mitigatory actions established in any analyzed accident in these MODES. Therefore, the removal of the freedom of movement verification in MODES 3 and 4 will not result in a significant increase in consequences of a previously evaluated accident.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to preserve the reactor protection criteria in those MODES in which the control rods were assumed to provide a mitigatory function. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety established by the control rods is provided by the ability to fully insert the control rods on a reactor trip. This feature will be retained in those MODES in which the control rods are assumed to serve a mitigatory function. However, in those MODES where the control rods are not assumed to provide a mitigatory function, the deletion of the freedom of movement surveillance requirement does not result in a degradation of any margin of safety that may be afforded by the control rods. Thus, in MODES 3 and 4, the removal of this surveillance requirement does not involve a significant reduction in a margin of safety.

3.1 L13

NOT USED

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.1 L14

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

The removal of the shutdown action statements associated with non-compliance with the dissolved gas concentration limits does not result in any hardware changes, result in a physical alteration of the plant, or involve a change in the controls governing normal operation. Thus, the removal of these statements does not alter the assumed initiator of any evaluated accident, and hence, does not involve a significant increase in the probability of any previously evaluated accident. The relocation of the dissolved gas concentration requirements to the TRM will continue to ensure that appropriate limits and associated actions are established for proper operation of the control rod drive(s) and/or the control rod(s). Further, the plausible consequences associated with a failure to comply with the concentration limits will not result in the failure of the control rods to perform their intended safety function. Therefore, the removal of the shutdown action statements will not result in a significant increase in consequences of a previously evaluated accident.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to preserve the operating restrictions on reactor coolant dissolved gas concentrations. The proposed change will continue to impose actions to mitigate the consequences of the out-of-limit condition. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety established by the dissolved gas concentration limits will be preserved by the requirements relocated to the Technical Requirements Manual. Appropriate remedial actions will continue to be provided should an out-of-limit condition develop. The operating restrictions provide protection for the control rod drive(s) and/or control rod(s) should a reactor trip occur while the control rod drive pressure boundary housing was filled by non-condensable gas. However, the credible damage mechanisms to the control rod drive(s) and/or control rod(s) do not affect the ability of the control rods to perform their intended safety function. Thus, the removal of the shutdown actions does not involve a significant reduction in a margin of safety.

ITS DISCUSSION OF DIFFERENCES
ITS Section 3.1: Reactivity Control Systems

1 NUREG 3.1.1, 3.1.4, 3.1.5, 3.1.8, & 3.1.9 - Incorporated TSTF-009.

2 NUREG 3.1.8 & 3.1.9 - The Frequency of NUREG SR 3.1.8.3 (ITS SR 3.1.8.2) and NUREG SR 3.1.9.2. (ITS SR 3.1.9.2) was changed to specify "Prior to performance of PHYSICS TESTS." This Frequency requires the nuclear overpower trip setpoint be verified prior to the onset of PHYSICS TESTS which ensures that the established LCO conditions are satisfied, with respect to the trip function. The requirement to perform these NUREG SRs with a Frequency of 8 hours is excessively restrictive and unduly burdensome on the operation of the unit. The short time frame in which the unit is expected to be conducting PHYSICS TESTS requiring the exception to one or more LCOs does not warrant the increased verification requirements. Further, these SRs provide a verification of RPS system performance at a Frequency significantly shorter than that required of the RPS when operating in MODE 1 at RATED THERMAL POWER (ref. NUREG 3.3.1). No basis exists to imply that the RPS trip function, or its calibration, would behave differently than that observed during power operation. The Bases were changed to reflect this Frequency. This change is consistent with Generic Change TSTF-344.

3 NUREG 3.1.4 - Incorporated TSTF-143.

4 NUREG 3.1.3 - The Moderator Temperature Coefficient (MTC) limits in ITS 3.1.3 were modified to specify the current license requirements as presented in CTS 3.1.7.1. Because there is no MTC value presently specified in the ANO-1 COLR, nor is there a value to be relocated to the COLR, ITS 3.1.3 was revised to specify that the MTC shall be non-positive whenever THERMAL POWER is greater than or equal to 95% of RTP and shall be less positive than 0.9×10^{-4} Δk/k/°F whenever THERMAL POWER is less than 95% RTP. These changes are in accordance with current license basis. Further, this change results in ITS 3.1.3 establishing a maximum positive limit that is consistent with NUREG-1430.

In SR 3.1.3.1, the phrase "within the upper limit specified in the COLR" was changed to "within the limits" to coincide with the LCO requirements.

SR 3.1.3.2 has been deleted because the CTS contains no lower limit on MTC. The lower limit for MTC will remain under licensee administrative control. This value is validated through observation of core physics parameters over the cycle duration. These parameters have historically indicated close agreement between core design assumptions and actual core parameters thus indicating agreement between the actual MTC values and those assumed in the cycle reload analyses. These changes are consistent with current license basis.

The Bases for 3.1.3 were similarly modified to reflect the above described changes. In addition, the 3.1.3 LCO Bases were modified to include CTS Bases guidance that the positive MTC limit below 95% RTP is to be corrected to the 95% RTP power level. This results in a linearly decreasing positive MTC value as power is increased from Hot Zero Power to 95% RTP. This change is consistent with current license basis.

ITS DISCUSSION OF DIFFERENCES

5 NUREG 3.1.4 - The terms "trippable," "trippability" and "untrippable" as they relate to CONTROL RODS have been removed from several locations within ITS 3.1.4 and the supporting BASES. This change preserves the current license basis. The CTS does not distinguish between trippable and untrippable inoperable CONTROL RODS. ANO-1 will maintain its current approach of dealing with inoperable rods similarly whether trippable or untrippable. The deletion of the words "trippable," "trippability" and "untrippable" is consistent with CTS and represents no change in intent or application from current license basis.

NUREG ACTION D was deleted because ITS 3.1.4 ACTIONS A, B and C, with the indicated changes, provide the requirements for all inoperable CONTROL RODS. Inoperable CONTROL RODS will continue to be dealt with consistently whether "trippable" or "untrippable." This maintains requirements consistent with CTS.

These changes are acceptable because the negative reactivity worth of an untrippable CONTROL ROD can be easily compensated for in the SHUTDOWN MARGIN (SDM) verification. SDM verification is the first Required Action in ITS 3.1.4. Thus, core reactivity and SDM considerations during operation are preserved in accordance with safety analysis assumptions. Further, if the CONTROL ROD is aligned within limits of its group average position (and the group average position is within the limits of ITS 3.2.1), then the power distribution of the core is unaffected. This similarly preserves the initial power distribution conditions of the safety analysis. Therefore, ITS Conditions A, B and C provide appropriate actions for continued operation with either an untrippable CONTROL ROD or an otherwise trippable CONTROL ROD that has been declared inoperable for some other reason.

The Bases have been revised to be consistent with the above mentioned changes. In addition, the Bases for SR 3.1.4.3 were modified to include additional detail regarding the control rod drop time testing. This change is consistent with current license basis.

6 NUREG 3.1.2, 3.1.3, & 3.1.4 - The word "Once" has been added to the Frequency of SR 3.1.2.1, SR 3.1.3.1, and SR 3.1.4.3 in ITS Section 3.1. This addition has been made to provide consistency between this statement of Frequency and the information contained within NUREG Section 1.4, Frequency. Discussions within Section 1.4 repeatedly emphasize the use of the term "Once" in this type of statement of Frequency. This change has been made specifically for clarification and consistency, and is considered to be editorial.

7 NUREG 3.1.6 - The wording of ITS 3.1.6 LCO and Condition A was changed to be consistent with the statements presented in ITS 3.1.4 LCO and Condition A. This editorial change establishes consistency between similar LCOs within the ITS.

ITS 3.1.6 Applicability will be MODES 1 and 2 in accordance with TSTF-159, Rev1.

NUREG 3.1.6 Required A.2 was replaced by the requirement to reduce THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER with a Completion Time of 2 hours. This required power reduction ensures that local LHR increases, due to a misaligned APSR, will not cause the core design criteria to be exceeded. This change incorporates an action that is implied by the current license basis.

The Bases were revised as necessary to reflect these changes. In addition, the last paragraph of the LCO Bases was revised to remove reference to peaking factors, leaving reference only to LHRs. No change in intent is associated with this change which is consistent with changes made elsewhere in the ITS Bases.

8 NUREG 3.1.4 - A portion of the methodology specified in NUREG SR 3.1.4.2 has been deleted. This change was made to maintain testing requirements consistent with the CTS. The CTS does not contain this level of detail with regard to CONTROL ROD testing. Specific methodology, including the minimum distance a CONTROL ROD must be moved during testing, is currently contained in documents under licensee control and for consistency will be maintained under licensee control. Removal of these details will not change the intent of the SR and will maintain current testing requirements.

Further, to maintain consistency with the NUREG Bases, the words "by moving" were replaced with the word "for." This change takes into account that more than one method of determining rod freedom or the basis for the inability to demonstrate movement of a CONTROL ROD exists. These changes preserve the intent of this SR which is to insure that the CONTROL RODS are capable of inserting into the core in the event of a reactor trip. Moreover, the NUREG SR 3.1.4.2 Bases attempt to establish exceptions to the SR which requires the freedom of movement be demonstrated "by moving." The Bases allow a determination of trippability that may be used to preserve CONTROL ROD OPERABILITY although the CONTROL ROD may not be capable of being moved. This constitutes an SR 3.0.1 exception established within the Bases which is inappropriate.

9 NUREG 3.1.4 - ITS SR 3.1.4.3 was modified to maintain CONTROL ROD drop time testing consistent with CTS 4.7.1.1 requirements. This change does not add new requirements nor does it change or remove any existing requirements.

The NOTE in NUREG SR 3.1.4.3 was modified to allow continued operation with reactor coolant pump combinations which provide less total reactor coolant system flow than the combination used during CONTROL ROD drop time testing. Continued operation is allowed provided the total reactor coolant flow is less than the total flow during testing. This allowance is appropriate due to the bounding nature of the test flow conditions. ANO-1 is currently licensed for limited operation in a one RCP per loop configuration. This change will allow for continued unit operation, to the extent allowed by CTS 3.1.1.1.A. Without this change to the Note, reducing the number of running RCPs from 3 to 2, with drop time testing having been performed with 3 RCPs running, would have required that all CONTROL RODS be declared inoperable. This declaration is unnecessarily restrictive due to the bounding nature of the test flow conditions.

A portion of the NUREG Bases for SR 3.1.4.3 was deleted because it established a condition requiring performance of the SR that was not consistent with the SR Frequency requirements. The ITS SR Frequency is given as "once prior to reactor criticality after each removal of the reactor vessel head." However, the Bases stated that the SR is required "after CONTROL ROD drive system maintenance or modification." This Bases condition is not included within the scope of the SR 3.1.4.3 Frequency and was therefore deleted.

10 NUREG 3.1.4 - Required Action A.2.3 has been shown as not adopted in the ITS. This item was not a requirement in the CTS for this Condition. The Required Action's reduction of the nuclear overpower trip setpoint does not actively contribute toward the mitigation of the negative effects of operation with a misaligned CONTROL ROD. This type of administrative action is better suited as a licensee controlled procedural action. Lastly, the Bases implication that this reduction in setpoint maintains core protection and operating margins is not supported. By not adopting this Required Action, requirements consistent with current license bases are being maintained.

BASES information for this Required Action has likewise been removed.

11 NUREG 3.1.4 - The Completion Time for Required Action A.1 (ITS 3.1.4 Required Action A.2.1) has been changed from 1 hour to 2 hours. The Required Action of realigning a misaligned CONTROL ROD is not specified in CTS. There is an implied Action presented by CTS 3.5.2.2.6. This specification allows for continued operation above 60% ALLOWABLE THERMAL POWER (ATP) if a previously misaligned CONTROL ROD is no longer misaligned. No Completion Time is specified for either this Specification or CTS 3.5.2.2.5 which requires the power reduction to less than 60% ATP. Due to the lack of current specified Completion Times for the Required Actions of reducing power to less than 60% ATP and realigning a misaligned CONTROL ROD, similar Completion Times of 2 hours have been adopted for both Required Actions. This 2 hour Completion Time along with ITS 3.1.4 Required Action A.1.1 ensures that, within 1 hour, proper SDM is verified or appropriate actions initiated, and within 2 hours, any misaligned CONTROL ROD is realigned or power is reduced below 60% ATP.

12 NUREG 3.1.4, 3.1.6, & 3.1.7 - Incorporated TSTF-110, Rev 2.

13 NUREG 3.1.8 & 3.1.9 - Incorporated TSTF-154, Rev 2. This generic change has been modified to reference the criterion of 10CFR50.36 instead of the NRC Policy Statement. This is an editorial change associated with implementation of the 10CFR50.36 rule changes after NUREG-1430, Rev 1 was issued.

14 NUREG 3.1.5 - Incorporated TSTF-216.

15 NUREG 3.1.8 & 3.1.9 - NUREG LCO 3.1.8 and LCO 3.1.9 were modified to include the allowance to suspend the requirements of ITS 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," during PHYSICS TESTS in MODES 1 and 2. The inclusion of this exception in the ITS is acceptable based on approved written procedures, administrative controls, the requirements of 10CFR50.59, and ITS LCO 3.1.8 and LCO 3.1.9 provisions in effect during the conduct of PHYSICS TESTS. This exception accommodates LCO 3.2.2 suspension that may be necessary to verify the fundamental characteristics of the nuclear reactor which is critical in demonstrating the adequacy of design, analytical models, and confirmation of analysis results. This change maintains requirements consistent with CTS 3.5.2.5.4.

Required changes to the Bases of ITS 3.1.8 and 3.1.9 were also made. An insert to the Bases was made to further clarify the basis for the acceptability of allowing PHYSICS TESTS exceptions. This Bases addition is entirely editorial in nature. Reference to Regulatory Guide 1.68, Revision 2, August 1978, and ANSI/ANS-19.6.1-1985, December 13, 1985, were deleted at each occurrence and replaced with reference to SAR Section 3A.9, "Startup Program - Physics Testing." ANO is not committed to Regulatory Guide 1.68 or ANSI/ANS-19.6.1. This change is consistent with current license basis.

16 NUREG 3.1.7 - The LCO, Actions and Note have been modified to maintain requirements consistent with the CTS requirements for CONTROL ROD and APSR position indication channel requirements. CTS 4.7.1.3 requires only one OPERABLE channel of position indication per rod. If this required channel is inoperable, the associated rod must be declared inoperable and the Actions of the rod's governing Specification must be completed. The CTS requirements are maintained by the indicated changes to ITS 3.1.7.

SR 3.1.7.1 was modified to match the requirements of ITS 3.1.7. This change was made to provide for Surveillance Requirements which adequately address the equipment required by the LCO. This change provides clarification of the inconsistency within the CTS with regard to the required channels of position indication and surveillance requirements. CTS Table 4.1-1, Items 23 and 24 required shiftly checks of both the absolute and relative rod position indication channels, while CTS 4.7.1.3 allowed for unrestricted operation with either or potentially both of these channels inoperable. This change ensures that only the channel which is being credited as providing the required indication need be checked.

ITS SR 3.1.7.2 was also added. This addition maintains testing requirements and Frequency consistent with CTS Table 4.1-1, Items 23 and 24.

17 NUREG 3.1.4 - The Required Actions for ITS 3.1.4 Condition A were reordered. This change was made due to the fact that inoperable and misaligned CONTROL RODS, whether trippable or not, are dealt with similarly by CTS and ITS (Reference DOD 5). Without this change in the order of the Required Actions, verification of proper SDM would not be required during operation with an inoperable (potentially untrippable) rod if it was aligned within 6.5% of its group average height as stipulated in NUREG Required Action A.1. The failure to verify adequate SDM is inappropriate in this condition. This change maintains requirements consistent with CTS requirements. Supporting changes to the order and content of BASES information were also made.

18 NUREG Bases - The Criterion statement at the conclusion of the Applicable Safety Analysis section was modified at each occurrence to refer to 10CFR50.36 instead of the NRC Policy Statement. This is an editorial change associated with the implementation of the 10CFR50.36 rule changes after NUREG-1430, Revision 1 was issued.

For ITS LCOs 3.1.1, 3.1.3, 3.1.4, 3.1.5, and 3.1.7, the 10CFR50.36 Criterion satisfied by the respective ITS LCOs was modified to preserve consistency with the ANO-1 license basis. Specifically, ANO-1 safety analyses upon which ITS LCOs 3.1.1, 3.1.3, 3.1.4, 3.1.5, and 3.1.7 are based were performed with the reactor critical. The ITS Applicability for these Specifications will be MODES 1 and 2. Thus, the Criterion statement was revised to specify that the LCO parameter satisfies Criterion 2 of 10CFR50.36 when in MODES 1 and 2 while critical. When in MODE 2 with the reactor subcritical, the LCO parameter satisfies Criterion 4 of 10CFR50.36. This change is consistent with current license basis and 10CFR50.36.

19 NUREG Bases 3.1.4 - The Bases for ITS 3.1.4 were modified to refer to a Linear Heat Rate (LHR) verification rather than a power peaking factor verification. These changes are consistent with the Bases discussion for ITS 3.2.5, "Power Peaking." Although LHR will be specified, no change in intent is associated with these changes. This is true because LHR verification is direct confirmation using the incore detector system that the core is operating within the design thermal operating limits. For additional information regarding this change, refer to Section 3.2 DOD 31.

20 NUREG 3.1.8 - Item c of the LCO requirements for maintaining the Nuclear Heat Flux Hot Channel Factor and the Nuclear Enthalpy Rise Hot Channel Factor within the limits specified in the COLR was modified in the ITS to specify that the linear heat rate (LHR) be maintained within the limits specified in the COLR. This change is necessary to provide PHYSICS TESTS requirements that are consistent with ITS 3.2.5, "Power Peaking" requirements. This LCO 3.1.8 condition coupled with SR 3.1.8.2 provides acceptable assurance that excessive core LHRs will not exist such that the thermal design limits of the fuel are exceeded. Although the terminology is different, this LCO condition preserves operating restrictions during PHYSICS TESTS consistent with those established in NUREG-1430.

In addition to the terminology change, a Note was added to the LCO, Condition B and SR 3.1.8.2 that specifies that the LCO provision on LHR only applies when THERMAL POWER is greater than 20% RTP. This Note establishes consistency between the LCO provisions of ITS 3.1.8 and ITS 3.2.5. This change is consistent with TSTF-160, Rev 1.

The Bases for ITS 3.1.8 were revised to reflect these changes.

21 NUREG Bases 3.1.3 - Repeated reference to SAR Chapter 14 using multiple reference indications is unnecessary and duplicative. Adequate reference to the SAR is provided by the first words of the introduction into the Applicable Safety Analyses portion of this Bases section.

22 NUREG Bases 3.1.2 - The Bases for ITS 3.1.2 were rewritten in their entirety to reflect the unit specific methodology of performing the reactivity anomaly determination. The NUREG Bases discussion centered around a comparison of the RCS boron concentration with a critical boron concentration curve (boron rundown curve) derived as part of the reload analyses. The ITS was written to reflect that ANO-1 performs a reactivity balance and then compares the value against a known reactivity condition (i.e., net reactivity of zero condition when the reactor is critical). Under critical conditions, a calculated net reactivity of a value other than zero would indicate the existence of a discrepancy in the reactivity parameters used in the calculation. This would then have to be evaluated in accordance with the discussion that was present in the NUREG Bases for LCO 3.1.2.

23 NUREG 3.1.9 - CTS 3.1.8.1 requires that the nuclear overpower trip be set at less than or equal to 5% RTP during the conduct of low power PHYSICS TESTS. Therefore, ITS 3.1.9 and SR 3.1.9.2 will specify that the Nuclear Overpower Trip Setpoint be set at 5% RTP rather than the 25% RTP value established by NUREG-1430. In addition, ITS 3.1.9.b was editorially modified to use terminology consistent with ITS 3.1.8.b and other locations in NUREG-1430. Specifically, ITS 3.1.9.b was modified to read that the "Nuclear overpower trip setpoint is set to ≤5% RTP."

24 NUREG 3.1.9 - The Applicability was modified to read as "During PHYSICS TESTS initiated in MODE 2." This Applicability is required in order to ensure that the Required Action A.1 is completed should THERMAL POWER exceed 5% RTP. As presently written in NUREG-1430, upon exceeding 5% RTP the unit is in MODE 1 and the LCO and its requirements no longer apply. This change is consistent with TSTF-256.

25 NUREG 3.1.9 - Incorporated TSTF-156, Rev 1.

26 NUREG Bases 3.1.9 - Bases information designated in NUREG-1430 as being applicable to SR 3.1.9.1 has been removed because the SR described by this Bases information does not appear in NUREG-1430. The subsequent Bases discussions of SR 3.1.9.2 through SR 3.1.9.4 were renumbered as appropriate due to this deletion.

27 Not used.

28 NUREG Bases 3.1.1 - The Bases for 3.1.1, SDM, was rewritten in its entirety to address ANO-1 current license and administrative requirements. ANO-1 CTS did not establish a required SHUTDOWN MARGIN (SDM) in MODES 3, 4 and 5. ANO-1 is a "hot shutdown" unit in that no safety analyses have been performed in MODES 3, 4 and 5. SAR analyses performed demonstrate the ability of the unit to establish hot shutdown conditions from operating conditions. Thus, all reference to analyses protected by the LCO 3.1.1 requirement was deleted from the Bases. SAR requirements are that the reactor be sufficiently shutdown to preclude inadvertent criticality in the shutdown condition.

ANO-1 has administratively verified adequate SHUTDOWN MARGIN during MODES 3, 4 and 5. In this verification, appropriate credit has been given to withdrawn CONTROL RODS (cocked rod protection), RPS operating mode (interpreted as whether the RPS was in Shutdown Bypass mode) and potential reactivity effects associated with the current plant operating condition. The required degree of subcriticality is maintained through boration, as necessary.

SR 3.1.1.1 is the method of verification of adequate SDM and is referenced from numerous MODE 1 and 2 LCOs. As such, the information in the Bases must support the derivation of SDM in MODES 1 through 5. Thus, additional reactivity parameters associated with unit operation above the point of adding heat have been added. The specific methodology for performing a SDM calculation will be maintained under licensee administrative control.

29 NUREG 3.1.2 - Incorporated TSTF-142.

30 NUREG Bases 3.1.1 - Reference to a specific volumetric flow rate, a specific boron concentration and a specific differential boron worth in deriving an example for approximate boration duration is inappropriate. All of these factors are a function of system operating characteristics, limitations, time in core life or available boration source. The more appropriate method is to establish boration from an appropriate source and to maximize the injection to the extent possible with consideration for reactor coolant system inventory and makeup and letdown system capacities. Further, this boration is required to continue until the boron concentration is verified to be sufficient to achieve the required shutdown margin.

31 Not used.

32 NUREG Bases 3.1.2 - The NUREG Bases statement that ITS 3.1.2 does not apply in MODE 6 was modified to remove reference to post-criticality testing that verifies the SDM. The verification of SDM in MODE 2 is of little benefit in assuring adequate SDM in MODE 6. The statement that fuel loading continually changes the reactivity condition of the core is correct and a portion of the basis for the SDM requirements in MODE 6 as stated.

33 NUREG 3.1.4 - A Note was added to precede ITS 3.1.4 Required Action A.2.2.3 (NUREG 3.1.4 Required Action A.2.5) that specifies the performance of SR 3.2.5.1 for verification of core power distribution only applies when THERMAL POWER is greater than 20% RTP. This Note is necessary to establish a correlation between the minimum power level at which the incore detector system can be reliably used to provide accurate indication of core power distribution and when the SR is required to be performed. This Note establishes consistency between the Required Action and ITS 3.2.5. This change is consistent with TSTF-160, Rev 1.

The Bases were similarly modified to include the Note.

34 NUREG Bases 3.1.6 - The Applicable Safety Analysis discussion for ITS 3.1.6 is revised to reflect ANO plant specific design and analysis. There are no explicit safety analyses associated with misaligned APSRs. Limits on their alignment are specified in the ITS to preserve assumptions used in the power distribution analysis that supports ITS LCO 3.2.1, LCO 3.2.3 and LCO 3.2.4. This change is consistent with current license basis.

ITS DISCUSSION OF DIFFERENCES

35 NUREG Bases 3.1.4 - The entire discussion of a second type of CONTROL ROD misalignment was deleted from the Bases. The NUREG Bases identified a second type of misalignment associated with a failure of one CONTROL ROD to insert (i.e. remain fully withdrawn) while all other CONTROL RODS insert fully. This discussion is inappropriate for the Bases of an LCO having Applicability in MODES 1 and 2 because: 1) the misalignment does not result in power peaking such that thermal design limits of the fuel would be exceeded, and 2) the misalignment is already discussed and provided for in the Bases for LCO 3.1.1, "Shutdown Margin (SDM)."

36 NUREG Bases 3.1.6 - The indicated changes remove all reference to a dropped APSR. The APSR mechanical design precludes its dropping into the reactor should its associated Control Rod Drive Mechanism become deenergized. It is non-credible for an APSR to drop into the reactor or become misaligned from its group due to dropping. The removal of these sentences does not alter the intent of the remaining passages or the Specification.

37 NUREG Bases 3.1.4 - The indicated changes represent clarification of the logic associated with the relationship between the relative position indicator and the power supply to the CONTROL ROD drives. Individual rods and groups may receive power from their associated group power supply, DC hold power supply or from the auxiliary power supply (as appropriate). Different power supply alignments to individual rods within a group could result in variations in the relative position indication for the rods within the group. The intent of the Bases statements remain the same. This change reflects unit design characteristics and is consistent with the current license basis.

38 NUREG Bases - The NUREG statement concerning the GDC criteria is modified in the ITS to reference the current licensing basis description contained in SAR Section 1.4.

39 NUREG Bases 3.1.8 - NUREG SR 3.1.8.4 (ITS SR 3.1.8.3) material describing the verification of SDM was erroneous. The listing of reactivity effects included parameters supporting the derivation of the SDM while subcritical or while critical below the point of adding heat. Neither is the case during the MODE 1 Applicability established for LCO 3.1.8. The reactivity effects listing was altered to incorporate the Doppler defect associated with heating of the fuel, Moderator defect associated with the heating of the reactor coolant and removal of the isothermal temperature coefficient (ITC) and RCS average temperature. The paragraph describing the necessity of using the isothermal temperature coefficient because the reactor is subcritical is deleted because it is obviously wrong in MODE 1.

Similarly, NUREG SR 3.1.9.4 (ITS SR 3.1.9.3) material describing the verification of SDM was also erroneous. The listing of reactivity effects included parameters supporting the derivation of the SDM while subcritical or while critical below the point of adding heat but did not support derivation of SDM when operating above the point of adding heat. The reactivity effects listing was altered to incorporate the Doppler defect associated with heating of the fuel and Moderator defect associated with the heating of the reactor coolant. The paragraph describing the necessity of using the isothermal temperature coefficient because the reactor is subcritical was modified to reflect that critical conditions may also exist. This change is consistent with TSTF-249.

ITS DISCUSSION OF DIFFERENCES

40 NUREG 3.1.5 - Incorporated TSTF-158, Rev 1.

41 NUREG LCO 3.1.6 Condition B has not been incorporated in the proposed ITS. Once power is reduced below 60%, the Required Action of Condition A (RA A.2) would be satisfied, allowing the unit to exit from Condition B. Therefore, the Condition B Required Actions would never be completed and are not required. This change is consistent with the current license basis.

42 NUREG LCO 3.1.9 allows LCO 3.2.1 "restricted operation region only" requirements to be suspended during PHYSICS TESTS. This exception is modified in the ITS 3.1.9 to allow suspension of LCO 3.2.1 requirements, consistent with CTS provisions which allow exception to position limit (does not limit to regulating rods inserted in the restricted region only) and overlap and sequence limits. This is acceptable since limits on THERMAL POWER and shutdown capability maintained during the PHYSICS TESTS ensure fuel damage criteria are preserved even if an accident were to occur with the LCO suspended.

43 Additional information has been incorporated to clarify that the value provided in the ITS 3.1.4 and 3.1.6 LCOs account for all necessary uncertainties and that the implementing procedures are not required to account for any additional uncertainties. This is consistent with the interpretation of the current requirements associated with Control Rod and APSR misalignment.

44 The NUREG Bases 3.1.5 Applicable Safety Analysis discussion has been revised to properly characterize the ANO acceptance criteria for the safety and regulating rod group insertion limits and operability or misalignment. The SAR does not state the acceptance criteria that the core remains subcritical for this event. However, B & W has placed a design objective in the cycle reload methodologies that the core will remain subcritical. A reference to the B & W topical report has also been added. This change is consistent with the current license basis.

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N/A

1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 The SDM shall be ~~greater than or equal to~~ within the limit ~~specified~~ provided in the COLR. ~~The minimum limit shall be ≥ [1.0]% Δk/k.~~

APPLICABILITY: MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM greater than or equal to the limit specified in the COLR.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Balance

CTS

LCO 3.1.2 The measured core reactivity balance shall be within ± 1% $\Delta k/k$ of predicted values. 4.9

APPLICABILITY: MODES 1 and 2. 4.9

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Measured core reactivity balance not within limit.	A.1	Re-evaluate core design and safety analysis and determine that the reactor core is acceptable for continued operation.	~~72 hours~~ 7 days
	AND		
	A.2	Establish appropriate operating restrictions and SRs.	7 days ~~72 hours~~
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours

4.9

29

N/A

N/A

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
SR 3.1.2.1	-------------------NOTES------------------- 1. The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. 2. This Surveillance is not required to be performed prior to entry into MODE 2. --- Verify measured core reactivity balance is within ± 1% Δk/k of predicted values.	Once ⑥ Prior to entering MODE 1 after each fuel loading AND -----NOTE------ Only required after 60 EFPD --------------- 31 EFPD thereafter	NA 4.9 NA NA 4.9

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be ~~maintained within the limits specified in the COLR. The maximum positive limit shall be [≤ [ ] Δk/k/°F at RTP].~~ non-positive whenever THERMAL POWER is ≥ 95% RTP and shall be less positive than 0.9×10^{-4} Δk/k/°F whenever THERMAL POWER is < 95% RTP. 3.1.7.1

4

APPLICABILITY: MODES 1 and 2. 3.1.7.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. MTC not within limits.	A.1 Be in MODE 3.	6 hours	3.1.7.3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.1.3.1	Verify MTC is within the ~~upper~~ limits. ~~specified in the COLR.~~	Once ~~Prior~~ prior to entering MODE 1 after each fuel loading	3.1.7.2

6

4

~~(continued)~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.2	-------------------NOTES------------------- 1. This SR is not required to be performed prior to entry into MODE 1 or 2. 2. If the MTC is more negative than the COLR limit when extrapolated to the end of cycle, SR 3.1.3.2 may be repeated. Shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. --- Verify extrapolated MTC is within the lower limit specified in the COLR.	Each fuel cycle within 7 EFPDs after reaching an equilibrium boron concentration equivalent to 300 ppm

④

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 CONTROL ROD Group Alignment Limits

4.7.1.2

LCO 3.1.4 Each CONTROL ROD shall be OPERABLE and aligned to within ~~[6.5]%~~ 6.5% of its group average height.

APPLICABILITY: MODES 1 and 2.

3.5.2

ACTIONS

5

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~[trippable]~~ CONTROL ROD inoperable, or not aligned to within ~~[6.5]%~~ 6.5% of its group average height, or both.	A.2.1 ~~A.1~~ ~~Align all CONTROL RODS in the group to within [6.5]% of the group average height, while maintaining the rod insertion, group sequence, and group overlap limits in accordance with LCO 3.2.1, "Regulating Rod Insertion Limits."~~ Restore CONTROL ROD alignment.	~~2~~ hours 2
	OR	
	A.1.1 ~~A.2.1.1~~ Verify SDM ~~is ≥ [1]% Δk/k~~ to be within the limit provided in the COLR.	1 hour AND Once per 12 hours thereafter
	OR	
	A.1.2 ~~A.2.1.2~~ Initiate boration to restore SDM to within limit.	1 hour
	AND	
		(continued)

3.5.2.2.6
N/A

11

17

3

17

3.5.2.2.2
3.5.2.2.3

1

N/A

17

3.5.2.2.2
3.5.2.2.3

Note to Reviewers
See Insert A for clarification of format of ACTION A.

INSERT A - Reviewer Clarification - LCO 3.1.4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CONTROL ROD inoperable, or not aligned to within 6.5% of its group average height, or both.	A.1.1 Verify SDM to be within the limit provided in the COLR. OR	1 hour AND Once per 12 hours thereafter
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2.1 Restore CONTROL ROD alignment. OR	2 hours
	A.2.2.1 Reduce THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER. AND	2 hours
	A.2.2.2 Verify the potential ejected rod worth is within the assumptions of the rod ejection analysis. AND	72 hours
	A.2.2.3 -------NOTE------- Only required when THERMAL POWER is > 20% RTP. ---------------- Perform SR 3.2.5.1.	72 hours

Insert after page 3.1-6

CTS

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. (continued)	A.2.2.1	Reduce THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER.	2 hours	3.5.2.2.5 N/A 17
	~~AND~~			
	~~A.2.3~~	~~Reduce the nuclear overpower trip setpoint to ≤ 70% of the ALLOWABLE THERMAL POWER.~~	~~10 hours~~	10
	AND			
	A.2.2.2 ~~A.2.4~~	Verify the potential ejected rod worth is within the assumptions of the rod ejection analysis.	72 hours	17 3.5.2.3 N/A
<INSERT 3.1-7A>	AND			
	A.2.2.3 ~~A.2.5~~	Perform SR 3.2.5.1.	72 hours	33 17 N/A
B. Required Action and associated Completion Time for Condition A not met.	B.1	Be in MODE 3.	6 hours	3.5.2.2.3 N/A
C. More than one ~~trippable~~ CONTROL ROD inoperable, or not aligned within ~~[6.5]%~~ 6.5% of its group average height, or both.	C.1.1	Verify SDM ~~is~~ to be ~~≥ [1] % Δk/k~~ within the limit provided in the COLR.	1 hour	5 3.5.2.2.2 3.5.2.2.3 1
	OR			
			(continued)	

<INSERT 3.1-7A>

A.2.2.3 ------------Note---------------
Only required when THERMAL POWER is > 20% RTP.

CTS

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
C. (continued)	C.1.2	Initiate boration to restore SDM to within limit.	1 hour	3.5.2.2.2 3.5.2.2.3
	AND			
	C.2	Be in MODE 3.	6 hours	3.5.2.2.1
~~D. One or more rods untrippable.~~	~~D.1.1~~	~~Verify SDM ~~is~~ to be ~~≥ [1]% Δk/k~~ within the limit provided in the COLR~~ (1)	~~1 hour~~	(5)
	~~OR~~			
	~~D.1.2~~	~~Initiate boration to restore SDM to within limit.~~	~~1 hour~~	
	~~AND~~			
	~~D.2~~	~~Be in MODE 3.~~	~~6 hours~~	

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
SR 3.1.4.1	Verify individual CONTROL ROD positions are within ~~[6.5]%~~ 6.5% of their group average height.	~~4 hours when the asymmetric CONTROL ROD alarm is inoperable~~ ~~AND~~ 12 hours ~~when the asymmetric CONTROL ROD alarm is OPERABLE~~ (12)	N/A
SR 3.1.4.2	Verify CONTROL ROD freedom of movement for ~~(trippability)~~ ~~by moving~~ each individual CONTROL ROD that is not fully inserted ≥ 3% ~~in any direction.~~ (5) (8)	92 days	Table 4.1-2 Item 2
SR 3.1.4.3	---NOTE--- With rod drop times determined with at least one but less than four reactor coolant pumps operating, operation may proceed provided operation is restricted to the pump combination operating during the rod drop time determination or pump combinations providing less total reactor coolant flow. (9) ------ Verify the rod drop time for each CONTROL ROD, from the fully withdrawn position, is ≤ ~~[1.66]~~ 1.66 seconds from power interruption at the CONTROL ROD drive breakers to $\frac{3}{4}$ insertion (25% withdrawn position) with T_{avg} ≥ 525°F.	Once ~~Prior to~~ prior to reactor criticality after each removal of the reactor vessel head (6)	N/A; 4.7.1.1 Table 4.1-2 Item 1

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Safety Rod Insertion Limits

LCO 3.1.5 Each safety rod shall be fully withdrawn. 3.5.2.1 3.1.3.5

APPLICABILITY: MODES 1 and 2. 3.1.3.5 3.5.2

move

-----NOTE-----
~~This LCO is not applicable while performing~~ SR 3.1.4.2. 3.5.2.5.1
Not required for any safety rod inserted to perform 14

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One safety rod not fully withdrawn.	~~A.1 Withdraw the rod fully.~~ ~~OR~~	~~1 hour~~	40
	~~A.2.1.1~~ A.1.1 Verify SDM ~~is~~ ~~≥ 1% Δk/k~~ to be within the limit provided in the COLR. OR	1 hour	3.5.2.1 1
	~~A.2.1.2~~ A.1.2 Initiate boration to restore SDM to within limit. AND	1 hour	3.5.2.1
	~~A.2.2~~ A.2 Declare the rod inoperable.	1 hour	3.1.3.7

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	CTS
B. More than one safety rod not fully withdrawn.	B.1.1	Verify SDM ~~is~~ ~~≥ 1% Δk/k~~ to be within the limit provided in the COLR	1 hour	3.5.2.1
	OR			
	B.1.2	Initiate boration to restore SDM to within limit.	1 hour	3.5.2.1
	AND			
	B.2	Be in MODE 3.	6 hours	3.5.2.2.1

1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.1.5.1	Verify each safety rod is fully withdrawn.	12 hours	N/A

3.1 REACTIVITY CONTROL SYSTEMS

CTS

3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

(7)

LCO 3.1.6 Each APSR shall be OPERABLE and aligned ~~within~~ to ~~[6.5]%~~ of its group average height. 4.7.1.2

(7)

APPLICABILITY: MODES 1 and 2. ~~when the APSRs are not fully withdrawn.~~ 3.5.2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One APSR inoperable, or to not aligned within its ~~limits,~~ 6.5% of its group average height, or both.	A.1 ~~Align the APSR group to within [6.5]% of the inoperable or misaligned rod, while maintaining the APSR insertion limits in the COLR.~~ Restore APSR alignment	2 hours
	~~AND~~ OR	
	A.2 ~~Prevent movement of the APSR group, while the rod remains inoperable or misaligned.~~ Reduce THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER.	2 hours
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Be in MODE 3.~~	~~6 hours~~

(3) 3.5.2.2.6 N/A

(7) N/A

(41)

SURVEILLANCE REQUIREMENTS

CTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify position of each APSR is within ~~≤ 6.5%~~ of the group average height.	~~4 hours when the asymmetric CONTROL ROD alarm is inoperable~~ ~~AND~~ 12 hours ~~when the asymmetric CONTROL ROD alarm is OPERABLE~~

N/A

12

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Position Indicator Channels

LCO 3.1.7 One ~~The absolute position indicator channel and the relative~~ position indicator channel for each CONTROL ROD and APSR shall be OPERABLE. (16) N/A

APPLICABILITY: MODES 1 and 2. N/A

ACTIONS

---NOTE---

Separate Condition entry is allowed for each ~~inoperable position indicator channel~~ CONTROL ROD and APSR. (16) N/A

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. The required ~~relative~~ position indicator channel inoperable for one or more rods.	~~A.1~~	~~Determine the absolute position indicator channel for the rod(s) is OPERABLE.~~	~~8 hours~~ ~~AND~~ ~~Once per 8 hours thereafter~~
~~B. The absolute position indicator channel inoperable for one or more rods.~~	~~B.1.1~~	~~Determine position of the rods with inoperable absolute position indicator by actuating the affected rod's zone position reference indicators.~~ ~~AND~~	~~8 hours~~

4.7.1.3

(16)

(continued)

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.1.2 Determine rods with inoperable position indicators are maintained at the zone reference indicator position and within the limits specified in LCO 3.1.5, "Safety Rod Insertion Limit"; LCO 3.2.1, "Regulating Rod Insertion Limits"; or LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," as applicable.	8 hours AND Once per 8 hours thereafter
	OR	
	B.2.1 Place the control groups with nonindicating rods under manual control.	8 hours
	AND	
	B.2.2 Determine the position of the nonindicating rods indirectly with fixed incore instrumentation.	8 hours AND Once per 8 hours thereafter AND (continued)

16

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)		------NOTE------ Not applicable during first 8 hour period ------------- 1 hour after motion of nonindicating rods, which exceeds [15 inches] in one direction since the last determination of the rod's position
C. The absolute position indicator channel and the relative position indicator channel inoperable for one or more rods. OR Required Action and associated Completion Time not met.	A.1 ~~C.1~~ Declare the rod(s) inoperable.	Immediately

16

4.7.1.3

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	~~Verify the absolute position indicator channels and the relative position indicator channels agree within the limit specified in the COLR.~~ Perform CHANNEL CHECK of required position indicator channel.	~~4 hours when the asymmetric CONTROL ROD alarm is inoperable~~ ~~AND~~ ~~12 hours when the asymmetric CONTROL ROD alarm is OPERABLE~~ 12 16

Table 4.1-1 Items 23 & 24.

<INSERT 3.1-17 A SR 3.1.7.2> 16

Table 4.1-1 Items 23 & 24

<INSERT 3.1-17A>

SR 3.1.7.2	Perform CHANNEL CALIBRATION of required position indicator channel.	18 months

 CTS.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions—MODE 1

LCO 3.1.8 During the performance of ~~PHYSICS~~ Group TESTS, the requirements of edit N/A

LCO 3.1.4, "CONTROL ROD Alignment Limits"; 3.1.3.5
LCO 3.1.5, "Safety Rod Insertion Limits"; N/A
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits"; 3.5.2.5.2
LCO 3.2.1, "Regulating Rod Insertion Limits," for the restricted operation region only; 3.5.2.5.3 3.5.2.5.4
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits";
LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits"; and 3.5.2.6
LCO 3.2.4, "QUADRANT POWER TILT (QPT)" 3.5.2.4.1 3.5.2.4.2 3.5.2.4.3

15

may be suspended, provided:

a. THERMAL POWER is maintained ≤ 85% RTP; N/A

b. Nuclear overpower trip setpoint is ≤ 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP; N/A

<INSERT 3.1-18A>

c. Linear Heat Rate (LHR) is ~~$F_Q(Z)$ and $F^N_{\Delta H}$ are~~ maintained within the limits specified in the COLR; and 20 N/A

d. SDM is ~~≥ [1.0]% Δk/k~~ within the limit provided in the COLR. 3.1.8.3 1

APPLICABILITY: MODE 1 during PHYSICS TESTS. N/A

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. SDM not within limit.	A.1	Initiate boration to restore SDM to within limit.	15 minutes
	AND		
	A.2	Suspend PHYSICS TESTS exceptions.	1 hour

N/A

(continued)

<INSERT 3.1-18A>

c. ------------NOTE-------------
Only required when THERMAL POWER is > 20% RTP.

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. THERMAL POWER > 85% RTP. OR Nuclear overpower trip setpoint > 10% higher than PHYSICS TESTS power level. OR Nuclear overpower trip setpoint > 90% RTP. <INSERT 3.1-19A> OR LHR ~~$F_Q(Z)$ or $F^N_{\Delta H}$~~ not within limits.	B.1 Suspend PHYSICS TESTS exceptions.	1 hour

N/A

20

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.8.1 Verify THERMAL POWER is ≤ 85% RTP.	1 hour
<INSERT 3.1-19B> SR 3.1.8.2 Perform SR 3.2.5.1.	2 hours
SR 3.1.8.3 Verify nuclear overpower trip setpoint is ≤ 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP.	~~8 hours~~ Prior to performance of PHYSICS TESTS

N/A

20 N/A

2 N/A

(continued)

<INSERT 3.1-19A>

-----NOTE-----
Only required when
THERMAL POWER
is > 20% RTP.

<INSERT 3.1-19B>

-------------NOTE-------------
Only required when THERMAL
POWER is > 20% RTP.

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.1.8.4 Verify SDM ~~is ≥ [1.0]% Δk/k~~	24 hours

To be within the limit provided in the COLR

NA

1

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.9 PHYSICS TESTS Exceptions—MODE 2

LCO 3.1.9 During performance of PHYSICS TESTS, the requirements of

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)"; — N/A
LCO 3.1.4, "CONTROL ROD Group Alignment Limits"; — N/A
LCO 3.1.5, "Safety Rod Insertion Limits"; — 3.1.3.5
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits"; — N/A (12)
LCO 3.2.1, "Regulating Rod Insertion Limits," ~~for the restricted operation region only;~~ and — 3.5.2.5.2, 3.5.2.5.3
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"; — 3.5.2.5.4
LCO 3.4.2, "RCS Minimum Temperature for Criticality" — 3.1.3.1

may be suspended, provided: — (5)

a. THERMAL POWER is ≤ 5% RTP; — 3.1.8.1.A, 3.1.8.1B

b. ~~Reactor trip setpoints on the OPERABLE~~ nuclear overpower ~~channels are~~ trip setpoint is set to ≤ ~~25%~~ 5% RTP; — 3.1.8.1.A, 3.1.8.1.B (23)

c. Nuclear instrumentation/~~source range and intermediate range~~ high startup rate CONTROL ROD withdrawal inhibit is ~~are~~ OPERABLE; and — (25) 3.1.8.2

d. SDM is ~~≥ [1.0]% Δk/k~~ within the limit provided in the COLR. — 3.1.8.3 (1)

APPLICABILITY: ~~MODE 2~~ during PHYSICS TESTS initiated in MODE 2. — (D) (24) N/A

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER not within limit.	A.1 Open control rod drive trip breakers.	Immediately

N/A

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	
B. SDM not within limit.	B.1	Initiate boration to restore SDM to within limit.	15 minutes	N/A
	AND			
	B.2	Suspend PHYSICS TESTS exceptions.	1 hour	
C. Nuclear overpower trip setpoint is not within limit. OR Nuclear instrumentation ~~source and intermediate range~~ high startup rate CONTROL ROD withdrawal inhibit inoperable.	C.1	Suspend PHYSICS TESTS exceptions.	1 hour	N/A 25

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.1.9.1	Verify THERMAL POWER is ≤ 5% RTP.	1 hour	N/A
SR 3.1.9.2	Verify nuclear overpower trip setpoint is ≤ ~~25%~~ 5% RTP.	~~8 hours~~ Prior to performance of PHYSICS TESTS	2 N/A

(continued)

23

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.1.9.3 Verify SDM ~~is ≥ [1.0]% Δk/k.~~ to be within the limit provided in the COLR	24 hours

1

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. per GDC 26 (Ref. 1). In MODES 3, 4, and 5 SDM requirements provide ~~sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). In MODES 3, 4, and 5, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all safety and regulating rods, assuming the single CONTROL ROD assembly of highest reactivity worth is fully withdrawn.~~ <INSERT B3.1-1A> 28

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of ~~movable control assemblies~~ CONTROL RODS and soluble boric acid in the Reactor Coolant System (RCS). In MODES 1 and 2, the CONTROL RODS can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, for analyzed events initiated in MODES 1 and 2, the CONTROL RODS, together with the Chemical Addition and Makeup and Purification System, provide SDM during power operation and are capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn (Ref. 1). edit 28 edit edit

The Chemical Addition and Makeup and Purification System can compensate for fuel depletion, during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions (Ref. 1). edit edit

During ~~power~~ operation in MODES 1 and 2, SDM control is ensured by operating with the safety rods fully withdrawn (LCO 3.1.5, "Safety Rod Insertion Limits") and the regulating rods within the limits of LCO 3.2.1, "Regulating Rod Insertion Limits." In MODE 3, consideration must be given to the position of the safety rods and whether the RPS is in Shutdown Bypass in determining the required SDM. When the unit is in ~~the shutdown and refueling modes~~ MODES 3, 4, or 5, the SDM requirements are met by means of adjustments to the RCS boron concentration. ~~Adjusted SDM limits defined in the COLR preclude recriticality in the event of a main steam line break (MSLB) in MODE 3, 4, or 5 when high steam generator levels exist.~~ Shutdown boron concentration requirements assume the highest worth rod is stuck in the fully withdrawn position to account for a postulated inoperable CONTROL ROD prior to reactor shutdown. 28

(continued)

<INSERT 3.1-1A>

maintain the core subcritical during these conditions.

In MODES 1 and 2 while critical, SDM requirements are met by the worth of the withdrawn CONTROL RODS which provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and abnormalities. In MODE 2 while subcritical and in MODE 3, with all safety rods withdrawn and the RPS not in Shutdown Bypass, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all CONTROL RODS , assuming the single CONTROL ROD of highest reactivity worth is fully withdrawn. In MODES 3, 4, or 5, when all safety rods are not fully withdrawn or the RPS is in Shutdown Bypass, the SDM defines the degree of subcriticality required to be maintained, assuming the CONTROL ROD of highest reactivity worth is fully withdrawn.

For analyzed events in MODES 1 and 2 while critical,

BASES (continued)

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with assumption of the highest worth rod stuck out following a reactor trip. ⊢(28)

abnormalities

edit

In MODES 1 and 2 while critical

The acceptance criteria for SDM requirements are that specified acceptable fuel design limits are maintained. The SDM requirements must ensure that: ⊢(28)

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events; and

b. The reactivity transients associated with postulated accident conditions are controllable with acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ 280 cal/gm energy deposition for the rod ejection accident); ~~and~~.

abnormalities

edit

In MODES 3, 4, and 5, the SDM requirements must ensure that

~~c.~~ The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. ⊢(28)

~~The most limiting accident for the SDM requirements is based on an MSLB, as described in the accident analysis (Ref. 2).~~

~~In addition to the limiting MSLB transient, the SDM requirement must also protect against:~~

~~a. Inadvertent boron dilution;~~

~~b. An uncontrolled rod withdrawal from a subcritical or low power condition;~~

~~c. Startup of an inactive reactor coolant pump;~~

~~d. Rod ejection; and~~

~~e. Return to criticality if an MSLB occurs during high steam generator level operations in MODE 3, 4, or 5.~~

~~The basis for the shutdown requirement when high steam generator levels exist is the heat removal potential in the~~ ⊢(28)

(continued)

In MODE 2 while subcritical and in MODES 3, 4, and 5, SDM satisfies Criterion 4 of 10 CFR 50.36.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~secondary system fluid and the negative reactivity added via MTC. At any given initial primary system temperature and its associated secondary system pressure, the secondary system liquid levels can be equated to a final primary system temperature assuming the entire mass is boiled. The resulting RCS temperature determines the required SDM.~~ (28)

In MODES 1 and 2 while critical, SDM satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 1). (18)

LCO

~~Shutdown boron concentration requirements assume the highest worth rod is stuck in the fully withdrawn position to account for a postulated inoperable or untrippable rod prior to reactor shutdown.~~ (28)

In MODES 1 and 2, and in MODE 3 when all safety rods are fully withdrawn and the RPS is not in Shutdown Bypass, SDM is a core design condition that can be ensured through CONTROL ROD positioning (control) regulating and (shutdown) safety groups and through the soluble boron concentration. (28)

In MODE 3 when all safety rods are not fully withdrawn or the RPS is in Shutdown Bypass, and in MODES 4 and 5, SDM represents a required degree of subcriticality that assumes the highest reactivity worth CONTROL ROD is fully withdrawn.

~~The MSLB (Ref. 2) accident is the most limiting analysis that establishes the SDM value of the LCO.~~

~~For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100 limits (Ref. 3).~~

~~To compensate for the potential heat removal associated with an MSLB accident when high steam generator levels exist during secondary system chemistry control and steam generator cleaning, the initial SDM in the core must be adjusted. The figure in the COLR represents a series of initial conditions that ensure the core will remain subcritical following an MSLB accident from those conditions.~~ (28)

APPLICABILITY

In MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to ensure that the reactor remains subcritical. ~~meet the assumptions of the safety analysis discussed above. The figure in the COLR is used to define the SDM when high steam generator levels exist during secondary system chemistry control and steam generator cleaning in MODES 3, 4, and 5.~~ (28)

¶ In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5

(continued)

BASES

APPLICABILITY (continued)

and LCO 3.2.1. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met. ~~If the SDM is below the limit for the steam generator level and RCS temperature specified in the COLR, RCS boration must be continued until the limit specified in the COLR is met.~~ 28

edit

In the determination of the required combination of boration flow rate and boron source concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank (BAAT) or the borated water storage tank (BWST). The operator should borate with the best source available for the plant conditions.

addition

unit

edit

~~In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle, when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of [1]% Δk/k must be recovered and a boration flow rate is [ ] gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by [1]% Δk/k. These boration parameters of [ ] gpm and [ ] ppm represent typical values and are provided for the purpose of offering a specific example.~~ 30

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

The SDM is verified by performing a reactivity balance calculation, ~~considering the listed~~ reactivity effects: (28)

. The reactivity effects that are considered in the reactivity balance are dependent upon the operational MODE of the unit. In general, the reactivity balance includes the following

a. RCS boron concentration; edit

b. ~~Regulating rod~~ CONTROL ROD position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; ~~and~~

g. Isothermal temperature coefficient (ITC) (28)

;
h. Moderator temperature coefficient (MTC); and
i. Doppler defect

Using the ITC accounts for Doppler reactivity in this calculation ~~because~~ when the reactor is subcritical, or critical but below the point of adding heat (POAH), and the fuel temperature will be changing at the same rate as the RCS.

Using the MTC and Doppler defect accounts for the reactivity effects of power operation above the POAH.

The Frequency of 24 hours is based on the generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. ~~10 CFR 50, Appendix A,~~ SAR, Section 1.4, GDC 26. (38)
2. ~~F~~SAR, Chapter ~~[14]~~ 3 edit
3. 10 CFR ~~100, "Reactor Site Criteria."~~ 50.36. (18)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Balance

BASES

BACKGROUND

abnormalities

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and ~~anticipated operational occurrences~~. Therefore, the reactivity balance is used as a measure of the predicted ~~versus measured~~ core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that safety analyses of design basis transients and accidents remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, CONTROL ROD, or burnable poison worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity. These could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing the predicted ~~versus measured~~ core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations ~~(LCO 3.1.1, "SHUTDOWN MARGIN (SDM)")~~ in ensuring the reactor can be brought safely to cold, subcritical conditions.

edit

22

agreement between the

Core reactivity and the actual

the predicted

Core reactivity with the actual

The difference between the actual and predicted core reactivity is commonly referred to as a reactivity anomaly.

22

edit

22

When the reactor ~~core~~ is critical ~~or~~ in ~~normal power operation~~, a reactivity balance exists ~~and the net reactivity is zero~~. A comparison of predicted and ~~measured~~ reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers, producing zero net reactivity. ~~Excess reactivity can be inferred from the boron letdown curve (or critical boron curve), which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed, (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations, and that the~~

MODE 1 or 2

where

Core reactivity

the actual core

(referred to as the actual core reactivity state)

and the net reactivity is known to be zero

Soluble boron and

22

(continued)

BASES

BACKGROUND (continued)

~~calculational models used to generate the safety analysis are adequate.~~ 22

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel remaining from the previous cycle provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical ~~at RTP and moderator temperature~~, the excess positive reactivity of the fuel is compensated by burnable absorbers (BA) ~~(if any)~~, CONTROL RODS, APSRs, thermal feedback from the fuel and moderator, fission product ~~whatever neutron~~ poisons (mainly xenon and samarium) ~~are present in the fuel~~, epithermal energy neutron absorbers, neutron leakage, and the RCS Reactor Coolant System boron concentration.

During cycle operation ~~When the core is producing THERMAL POWER~~, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the primary method of compensating for the reduction in excess reactivity is through a reduction in the RCS boron concentration ~~is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated~~.

22

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are the establishment of the reactivity balance limit to ensure that unit ~~plant~~ operation is maintained within the assumptions of the safety analyses. edit

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation ~~(Ref. 2)~~ is, therefore, dependent upon an accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as CONTROL ROD withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity (Ref. 2). These accident analysis evaluations rely on computer codes which have been qualified against available test data, operating unit ~~plant~~ data, and analytical benchmarks. Monitoring the core reactivity balance ensures that the nuclear methods provide an accurate representation of the core reactivity. edit edit 22 edit edit

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

behavior and the ~~RCS boron concentration~~ requirements for reactivity control during ~~fuel depletion~~. the operating cycle — 22

the actual reactivity condition of the critical reactor

The comparison between ~~measured~~ and predicted initial core reactivity provides a normalization ~~for~~ the calculational models used to predict core reactivity. If the ~~measured and~~ predicted ~~RCS boron concentrations for identical~~ core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict ~~soluble boron~~ reactivity requirements may not be accurate. If reasonable agreement between ~~measured~~ and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations ~~in the measured boron concentration from the predicted boron letdown curve~~, which is developed during ~~fuel depletion~~, may be an indication that the calculational model is not adequate for ~~core burnups beyond BOC~~, or that an unexpected change in core conditions has occurred. — 22

the / of / n opportunity for the / Core reactivity and the actual Core reactivity at reference / the actual / predicted reactivity condition from the actual reactivity condition during the operating cycle / the operating cycle

The normalization of the predicted ~~RCS boron concentration~~ to the ~~measured~~ value is typically performed after reaching RTP following startup from a refueling outage, with the CONTROL RODS in their ~~normal~~ positions ~~for~~ ~~power operation~~. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated, as core conditions change during the cycle. — 22

reactivity parameters / actual reactivity / and APSRs / reference / and fission product poisons at their expected equilibrium concentrations / RCS temperature

Reactivity balance satisfies Criterion 2 of ~~the NRC Policy Statement~~. 10 CFR 50.36 (Ref. 3) — 18

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled, once the core design is fixed. During operation, therefore, the conditions of the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the ~~Design Basis Accident (DBA) and transient~~ accident analyses are no longer valid, or that the uncertainties in the nuclear design methodology are larger than expected. A limit on the reactivity of ± 1% Δk/k has been established, based on engineering judgment. A ± 1% Δk/k deviation in the predicted reactivity — 22

(continued)

the actual reactivity condition of the reactor

BASES

LCO (continued)

from ~~that predicted~~ is larger than expected for normal operation and should therefore be evaluated.

the predicted

actual reactivity

When ~~measured~~ core reactivity is within 1% Δk/k of the ~~predicted~~ value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. ~~Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.~~ 22

APPLICABILITY

In MODES 1 and 2 ~~during fuel cycle operation with $k_{eff} \geq 1$~~, the limits on core reactivity must be maintained ~~because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER~~. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. 22

the

balance

to ensure an acceptable SDM and continued adherence to the assumptions used in the accident analysis

This Specification does not apply in MODES 3, 4, and 5, because the reactor is shutdown and changes to core reactivity due to fuel depletion cannot occur.

the net reactivity condition of the reactor can not easily be determined and

In MODE 6, ~~fuel loading results in a continually changing core reactivity.~~ Boron concentration requirements (LCO 3.9.1, "Refueling Boron Concentration") ensure that fuel movements are performed within ~~the~~ bounds ~~of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement or CONTROL ROD replacement or shuffling)~~. edit 32

acceptable

ACTIONS

A.1 and A.2

the actual core reactivity

the

Should an anomaly develop between ~~measured~~ and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input ~~to~~ design 22

the

assumptions used in the core

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of ~~72 hours~~ 7 days is based on the low probability of ~~a DBA~~ an abnormality or accident occurring during this period, and allows sufficient time to assess the physical condition of the ~~reactor~~ core and complete the evaluation of the core design and safety analysis. 29 22 edit

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core reference conditions at the time of ~~RCS boron concentration sampling~~, then a recalculation of ~~the RCS boron concentration requirements~~ the reactivity balance may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the ~~boron letdown curve~~ appropriate reactivity parameter may be renormalized, and ~~power~~ operation in MODE 1 may continue. If operational restrictions or additional surveillance requirements are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined. 22 22

The required Completion Time of ~~72 hours~~ 7 days is adequate for preparing operating restrictions or surveillances that may be required to allow continued reactor operation. 29

B.1

If the core reactivity balance cannot be restored to within the ±1% Δk/k limit, the unit must be brought to a MODE in which the LCO does not apply. ~~To achieve this status~~ As a conservative measure, the unit must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then boration required by Required Action A.1 of LCO 3.1.1 would occur. The allowed 22

(continued)

BASES

ACTIONS

B.1 (continued)

Completion Time of 6 hours is reasonable, based on operating experience to reach the required unit conditions from ~~full power~~ conditions in an orderly manner and without challenging unit systems.

RTP

edit

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic ~~comparisons of measured and predicted RCS boron concentrations~~. The comparison is made considering that ~~other~~ core conditions are fixed or stable, including CONTROL ROD positions, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1, as an initial check on core conditions and design calculations at BOC. A Note is included in the SR to indicate that the normalization of predicted core reactivity to the measured value ~~must~~ take place within the first 60 effective full power days (EFPD) after each fuel loading. ~~This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations.~~ The required ~~subsequent~~ Frequency of 31 EFPD, following the initial 60 EFPD after entering MODE 1 is acceptable, based on the slow rate of core reactivity changes due to fuel depletion and the presence of other indicators (QPT, etc.) for prompt indication of an anomaly. Another Note is included in the SRs to indicate that the performance of the Surveillance is not required for entry into MODE 2.

a

reactivity balance calculation that compares the predicted core reactivity to the actual core reactivity condition (net reactivity of zero condition)

and APSR

once

reactivity

after each fuel loading

may

22

The 60 EFPD after entering MODE 1 allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations.

REFERENCES

1. ~~10 CFR 50, Appendix A~~ SAR, Section 1.4, GDC 26, GDC 28, and GDC 29. 38
2. ~~F~~SAR, Chapter ~~[14]~~ 3A and. edit
3. 10 CFR 50.36 18

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

associated

BACKGROUND

(so that in the power operating range the net affect of the prompt inherent nuclear feedback characteristic tends to compensate for a rapid increase in reactivity)

According to GDC 11 (Ref. 1), the reactor core and ~~its interaction with the~~ Reactor Coolant System (RCS) ~~must~~ be designed ~~for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.~~ 5ro — 38

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). ~~The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation.~~ Therefore, (with a negative MTC) a coolant temperature increase will cause a reactivity decrease. ~~so that the coolant temperature tends to return toward its initial value.~~ Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result. ~~The same characteristic is true when the MTC is positive and coolant temperature decreases occur.~~ edit edit edit

~~MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements.~~ Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less than (or equal to) zero when THERMAL POWER is 95% RTP or greater. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional burnable absorbers to yield an MTC at BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles ~~that are designed to achieve high burnups or that have changes to other characteristics~~ are evaluated to ensure the MTC does not ~~exceed the EOC limit.~~ become more negative than the value assumed in the safety analyses. edit — 4 edit edit

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Reference 2 contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are initial conditions in the safety analyses, and both values must be bounded. Values used in the analyses consider worst case conditions, such as very large soluble boron concentrations for overheating events, to ensure the accident results are bounding. ~~(Ref. 2)~~ EDIT. 21

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis ~~(Ref. 2)~~ and 21

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

Accidents that cause core overheating (either decreased heat removal or increased power production) must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the CONTROL ROD withdrawal transient from either zero or full THERMAL POWER. The limiting overheating event relative to plant response is based on the maximum difference between core power and steam generator heat removal during a transient. The most limiting event with respect to positive MTC is ~~a rod withdrawal accident from zero power, also referred to as a~~ the startup accident. ~~(Ref. 4)~~ EDIT. 21

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event that produces the most rapid cooldown of the RCS, and is therefore the most limiting event with respect to the negative MTC, is a steam line break (SLB) event. Following the reactor trip for the postulated EOC SLB event, the large moderator temperature reduction, combined with the large negative MTC, may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, a substantial fraction of core power ~~is~~ may be produced with all CONTROL ROD assemblies inserted, except the most reactive one. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core power may be produced through the effects of subcritical neutron multiplication. EDIT.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

MTC values are bounded in reload safety evaluations, assuming steady state conditions at BOC and EOC. ~~A near EOC measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload/design predictions.~~ 4

In MODES 1 and 2 while critical, MTC satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10CFR 50.36 (Ref. 3). 18

In MODE 2 while sub-critical, MTC satisfies Criterion 4 of 10 CFR 50.36.

LCO

LCO 3.1.3 requires the MTC to be within specified limits ~~of the COLR~~ 4 to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The LCO establishes a maximum positive value that can not be exceeded. The limit of +0.9E-4 ~~(%)~~ Δk/k~~%~~/°F (corrected to 95% RTP) on positive MTC, when THERMAL POWER is < 95% RTP, ensures that core overheating accidents will not violate the accident analysis assumptions. edit 4 The requirement for a ~~negative~~ non-positive MTC, when THERMAL POWER is ≥ 95% RTP, ensures that core operation will be stable. 4 ~~The negative MTC limit for EOC specified in the COLR ensures that core overcooling accidents will not violate the accident analysis assumptions.~~ 4

MTC is a core physics parameter determined by the fuel and fuel cycle design and cannot be ~~easily~~ directly controlled once the core design is fixed during operation, therefore, the LCO can only be ensured through measurement. edit The surveillance check~~s~~ at BOC ~~and EOC~~ on MTC provide~~s~~ confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met. 4

APPLICABILITY

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from ~~THERMAL POWER~~ power operation will not violate the design assumptions of the accident analysis. edit In MODE 2, the limits must also be maintained to ensure that startup and subcritical accidents, such as the uncontrolled CONTROL ROD ~~assembly~~ or group withdrawal, will not violate the assumptions of the accident analysis. edit In MODES 3, 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis

(continued)

BASES

APPLICABILITY (continued)

assumption are initiated from these MODES. However, the variation of MTC with temperature in MODES 3, 4, and 5 for DBAs initiated in MODES 1 and 2 is accounted for in the subject accident analysis. The variation of MTC with temperature assumed in the safety analysis, is accepted as valid once the BOC and middle of cycle measurements ~~are~~ is used for normalization. 4

ACTIONS

A.1

core physics parameter determined by

MTC is a ~~function of~~ the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis ~~bounds~~ assumptions. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, for reaching MODE 3 conditions from ~~full power conditions~~ RTP in an orderly manner and without challenging ~~plant~~ unit systems. EDIT.

EDIT.

EDIT.

EDIT.

SURVEILLANCE REQUIREMENTS

move

The ~~following two~~ SRs for measurement of the MTC at the beginning ~~and end~~ of each fuel cycle provides for confirmation of the limiting MTC values. The MTC changes slowly from most positive (least negative) to most negative value during fuel cycle operation, as the RCS boron concentration is reduced with fuel depletion. 4

SR 3.1.3.1

EDIT.

The requirement for measurement, prior to initial operation ~~above 5% RTP~~ in MODE 1, satisfies the confirmatory check on the most positive (least negative) MTC value. EDIT.

~~SR 3.1.3.2~~

~~The requirement for measurement, within 7 effective full power days (EFPD) after reaching an equilibrium boron concentration of 300 ppm for RTP, satisfies the confirmatory~~ 4

~~(continued)~~ (continued) EDIT.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.2 (continued)

check on the most negative (least positive) MTC value. The measurement is performed at any THERMAL POWER equivalent to an RCS boron concentration of 300 ppm (for steady state operation at RTP with all CONTROL RODS fully withdrawn) so that the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. MTC values are extrapolated and compensated to permit direct comparison to the specified MTC limits.

4

The SR is modified by two Notes. Note 1 indicates performance of SR 3.1.3.2 is not required prior to entering MODE 1 or 2. Although this Surveillance is applicable in MODES 1 and 2, the reactor must be critical before the Surveillance can be completed. Therefore, entry into the applicable MODE, prior to accomplishing the Surveillance, is necessary.

Note 2 indicates that SR 3.1.3.2 may be repeated, and shutdown must occur, prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. The minimum allowable boron concentration is obtained from the EOC MTC versus boron concentration slope with appropriate conservatisms. Thus, the projected EOC MTC is evaluated before the lower limit is actually reached.

4

REFERENCES

1. SAR, Section 1.4 ~~10 CFR 50, Appendix A,~~ GDC 11. 38

2. ~~F~~SAR, Chapter ~~[14]~~ 5A and 14.

~~3. FSAR, Section [ ].~~ 21

EDIT.

3. ~~4.~~ ~~FSAR, Section [ ].~~ 10CFR50.36. 19

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 CONTROL ROD Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY ~~(e.g., trippability)~~ of the CONTROL RODS ~~(safety rods and regulating rods)~~ is an initial condition assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial condition assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. 5 EDIT. EDIT.

The applicable criteria for these design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Control System Redundancy-and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2). H 38

Mechanical or electrical failures may cause a CONTROL ROD to become inoperable or to become misaligned from its group. CONTROL ROD inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod CONTROL ROD worth for reactor shutdown. Therefore, CONTROL ROD alignment and OPERABILITY are related to core operation within design power peaking limits and the core design requirement of a minimum SDM. EDIT.

Limits on CONTROL ROD alignment and OPERABILITY have been established, and all rod CONTROL ROD positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved. EDIT.

CONTROL RODS are moved by their CONTROL ROD drive mechanisms (CRDMs). Each CRDM moves its rod ¾ inch for one revolution of the leadscrew, but at varying rates depending on the signal output from the Control Rod Drive Control System (CRDCS). 1b 1c EDIT.

The CONTROL RODS are arranged into rod groups that are radially symmetric. Therefore, movement of the CONTROL RODS does not introduce radial asymmetries in the core power distribution. The ~~safety rods and the regulating rods~~ CONTROL RODS EDIT.

(continued)

BASES

BACKGROUND (continued)

provide required negative reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating rods provide reactivity ~~(power level)~~ control during normal operation and transients, and their movement is normally controlled in ~~governed by the~~ automatic, control system. edit edit edit edit

The axial position of the CONTROL RODS ~~safety rods and regulating rods~~ is indicated by three ~~two separate and~~ independent systems, which are the relative position indicators ~~transducers~~ ~~and~~, the absolute position indicators ~~transducers~~, and the zone reference indicators (see LCO 3.1.7, "Position Indicator Channels"). edit edit

The relative position indicator transducer is a potentiometer that is driven by electrical pulses from the CRDCS. There is one counter for each CONTROL ROD drive. Individual rods in a group, when aligned to the same power supply, all receive the same signal to move; therefore, the counters for all rods in a group should normally indicate the same position. The Relative Position Indicator System is considered highly precise. ~~(one rotation of the leadscrew is 1 inch in rod motion).~~ ~~If~~ However, if a rod does not move for each demand pulse, the counter will still count the pulse and incorrectly reflect the position of the rod. edit 37

The Absolute Position Indicator System provides a highly accurate indication of actual CONTROL ROD position, but at a lower precision than the relative position indicators. This system is based on ~~inductive analog~~ signals from a series of reed switches spaced along a tube. ~~with a center to center distance of 3.75 inches.~~ edit edit edit

<INSERT B 3.1-18 A>> edit

APPLICABLE SAFETY ANALYSES

CONTROL ROD misalignment and inoperability accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing CONTROL ROD inoperability or misalignment are that:

a. There shall be no violations of:

    1. specified acceptable fuel design limits, or
    2. Reactor Coolant System (RCS) pressure boundary integrity ~~damage~~; and edit

b. The core must remain subcritical after an abnormality or accident. ~~transients.~~ edit

(continued)

<INSERT B3.1-18A>

Other reed switches included in the same tube with the absolute position indicator matrix provide full in and full out limit indications, and position indications at 0%, 25%, 50%, 75%, and 100% travel. This series of seven indicators are called zone reference indicators.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Two ~~Three~~ types of misalignment are distinguished. During MODES 1 and 2. During movement of a CONTROL ROD group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. ~~The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the CONTROL RODS to meet the SDM requirement with the maximum worth rod stuck fully withdrawn. If a CONTROL ROD is stuck in the fully withdrawn position, its worth is accounted for in the calculation of SDM, since the safety analysis does not take two stuck rods into account.~~ 35 The Second ~~third~~ type of misalignment occurs when one CONTROL ROD ~~rod~~ drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return towards the original power due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local linear heat rates (LHRs).

The accident analysis and reload safety evaluations define regulating rod insertion limits that ensure the required SDM can always be achieved if the maximum worth CONTROL ROD is stuck fully withdrawn (Ref. 3 ~~4~~). If a CONTROL ROD is stuck in or dropped in, continued operation is permitted if the increase in local LHR is within the design limits. The Required Action statements in the LCOs provide conservative reductions in THERMAL POWER and verification of SDM to ensure continued operation remains within the bounds of the safety analysis (Ref. 3 ~~5~~). EDIT. EDIT.

Continued operation of the reactor with a misaligned or dropped CONTROL ROD is allowed if the local core LHRs ~~$F_Q(Z)$ and the $F_{\Delta H}^N$~~ are verified to be within their limits in the COLR. When a CONTROL ROD is misaligned, the assumptions that are used to determine the regulating rod insertion limits, APSR insertion limits, AXIAL POWER IMBALANCE limits, and QPT limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and ~~$F_Q(Z)$ and $F_{\Delta H}^N$~~ must be verified directly by incore mapping. Bases Section 3.2, "Power Distribution Limits," contains a more complete discussion of the relation of LHR ~~$F_Q(Z)$ and $F_{\Delta H}^N$~~ to the operating limits. 19

(continued)

BASES

In MODES 1 and 2 while critical,

In MODE 2 while subcritical, the CONTROL ROD group alignment limits satisfy Criterion 4 of 10 CFR 50.36.

APPLICABLE SAFETY ANALYSES (continued)

the CONTROL ROD group alignment limits satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10CFR50.36 (Ref. 4). (18)

LCO

The limits on CONTROL ROD group alignment, safety rod insertion, and APSR alignment, together with the limits on regulating rod insertion, APSR insertion, AXIAL POWER IMBALANCE, and QPT, ensure the reactor will operate within the fuel design criteria. The Required Actions in these LCOs ensure that deviations from the alignment limits will either be corrected or that THERMAL POWER will be adjusted, so that excessive local LHRs will not occur and the requirements on SDM and ejected rod worth are preserved.

The limit for individual CONTROL ROD misalignment is ~~[6.5]%~~ 6.5% (approximately 9 inches) deviation from the group average position. This value is established, based on the distance between reed switches, with additional allowances for uncertainty in the absolute position indicator amplifiers, group ~~maximum or minimum synthesizer~~ average position calculator, and asymmetric alarm or fault detector outputs. Therefore no additional uncertainties are required to be incorporated in the implementing procedures. ¶ For the purpose of complying with this LCO the ~~The~~ position of ~~an inoperable~~ a misaligned rod is not included in the calculation of the rod group average position. EDIT. EDIT. (19)

Failure to meet the requirements of this LCO may produce unacceptable ~~power peaking factors and~~ LHRs, or unacceptable SDM or ejected rod worth, all of which may constitute initial conditions inconsistent with the safety analysis. (43)

APPLICABILITY

The requirements on CONTROL ROD OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which significant neutron (or fission) power is generated, and the OPERABILITY ~~(i.e., trippability)~~ and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the ~~CONTROL RODS are typically bottomed and the~~ reactor is shut down ~~and not producing fission power~~ and resultant local power peaking would not exceed fuel design limits. In ~~the shutdown MODES~~ 3, 4, 5 and 6, the OPERABILITY of the ~~safety and regulating rods~~ CONTROL RODS has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5, and (5) EDIT. EDIT. EDIT. EDIT. EDIT.

(continued)

BASES

APPLICABILITY (continued)

LCO 3.9.1, "Boron Concentration," for boron concentration requirements during ~~refueling~~ MODE 6.

EDIT.

ACTIONS

~~A.1~~ A.2.1

Alignment of the inoperable or misaligned CONTROL ROD may be accomplished by either moving the single CONTROL ROD to the group average position, or by moving the remainder of the group to the position of the single inoperable or misaligned CONTROL ROD. Either action can be used to restore the CONTROL RODS to a radially symmetric pattern. However, this must be done without violating the CONTROL ROD group sequence, overlap, and insertion limits of LCO 3.2.1, "Regulating Rod Insertion Limits," given in the COLR. THERMAL POWER must also be restricted, as necessary, to the value allowed by the insertion limits of LCO 3.2.1. The required Completion Time of ~~1 hour~~ 2 hours is acceptable because local xenon redistribution during this short interval will not cause a significant increase in LHR. This option of inserting the group to the position of the misaligned rod is not available if a safety rod is misaligned, since the limits of LCO 3.1.5, "Safety Rod Insertion Limits," would be violated.

MOVE DOWN TO FOLLOW A.1.2.

11

EDIT

~~A.2.1~~ A.1.1

Compliance with Required Actions of Condition A ~~A.2.1.1 through A.2.5~~ allows for continued power operation with one CONTROL ROD inoperable ~~but trippable~~, or misaligned from its group average position, or both. ~~These Required Actions comprise the final alternate for Condition A.~~

5

If realignment of the CONTROL ROD to the group average or alignment of the group to the misaligned CONTROL ROD is not completed within 1 hour ~~(Required Action A.1 not met)~~, the rod ~~should~~ shall be considered inoperable. Since the rod may be inserted farther than the group average insertion for a long time, SDM must be evaluated. Ensuring the SDM meets the minimum requirement established in the COLR within 1 hour is adequate to determine that further degradation of the SDM is not occurring.

move

move

17

EDIT.

(continued)

BASES

ACTIONS (continued)

~~A.2.1.2~~ A.1.2

If the SDM is less than the limit specified in the COLR, then the — EDIT.

Restoration of the required SDM requires increasing the RCS boron concentration, since the CONTROL ROD may remain misaligned and not be providing its normal negative reactivity on tripping. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour to initiate boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored. 1c 17

<INSERT A.2.1> from previous page 17

A.2.~~2~~.1

Reduction of THERMAL POWER to ≤ 60% ALLOWABLE THERMAL POWER ensures that local LHR increases, due to a misaligned rod, will not cause the core design criteria to be exceeded. The required Completion Time of 2 hours allows the operator sufficient time for reducing THERMAL POWER.

~~A.2.3~~

~~Reduction of the nuclear overpower trip setpoint to ≤ 70% ALLOWABLE THERMAL POWER, after THERMAL POWER has been reduced to 60% ALLOWABLE THERMAL POWER, maintains both core protection and an operating margin at reduced power similar to that at RTP. The required Completion Time of 10 hours allows the operator 8 additional hours after completion of the THERMAL POWER reduction in Required Action A.2.2 to adjust the trip setpoint.~~ 10

A.2.~~4~~2.2 17

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.65% Δk/k at RTP or 1.00% Δk/k at zero power (Ref. ~~6~~). This evaluation may require a computer calculation of the maximum ejected rod worth based on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the ~~remainder of the fuel cycle to ensure a valid~~ duration of time that operation is expected to continue with a misaligned rod. 3 EDIT. EDIT.

(continued)

BASES

ACTIONS A.2.2.2 (continued)

~~evaluation~~ should fuel cycle conditions at some later time become more bounding than those at the time of the rod misalignment. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation.

, additional evaluation will be required to verify the continued acceptability of operation.

A.2.2.3

Performance of SR 3.2.5.1 provides a determination of the ~~power peaking factors~~ using the Incore Detector System. Verification of the ~~$F_Q(Z)$ and $F_{\Delta H}^N$~~ from an incore power distribution map is necessary to ensure that excessive local LHRs will not occur due to CONTROL ROD misalignment. This is necessary because the assumption that all CONTROL RODS are aligned (used to determine the regulating rod insertion, AXIAL POWER IMBALANCE, and QPT limits) is not valid when the CONTROL RODS are not aligned. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and adequate time is allowed to obtain an incore power distribution map.

local core LHRs

<INSERT B3.1-23A>

B.1

If the Required Actions and associated Completion Times for Condition A ~~cannot be~~ are not met, the ~~plant~~ unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the ~~plant~~ unit must be brought to at least MODE 3 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from ~~full power conditions~~ RTP in an orderly manner and without challenging ~~plant~~ unit systems.

EDIT. EDIT. EDIT. EDIT.

C.1.1

More than one ~~trippable~~ CONTROL ROD becoming inoperable or misaligned, ~~or both inoperable but trippable and misaligned~~ or both, from their group average position, is not expected and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Ensuring the SDM meets the minimum requirement

17 17 19 33 5 EDIT.

(continued)

<INSERT B3.1-23A>

Required Action A.2.2.3 is modified by a Note that requires the performance of SR 3.2.5.1 only when THERMAL POWER is greater than 20% RTP. This establishes a Required Action that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

BASES

ACTIONS

C.1.1 (continued)

within 1 hour allows the operator adequate time to determine the SDM.

C.1.2

If the SDM is less than the limit specified in the COLR, then the ~~Restoration of the required SDM requires increasing~~ the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

EDIT.

C.2

If more than one ~~trippable~~ CONTROL ROD is inoperable or misaligned from their group average position, continued operation of the reactor may cause the misalignment to increase, as the regulating rods insert or withdraw to control reactivity. If the CONTROL ROD misalignment increases, local power peaking may also increase, and local LHRs will also increase if the reactor continues operation at THERMAL POWER. The SDM is decreased when one or more CONTROL RODS become inoperable at a given THERMAL POWER level, or if one or more CONTROL RODS become misaligned by insertion from the group average position.

(5)

Therefore, it is prudent to place the reactor in MODE 3. LCO 3.1.4 does not apply in MODE 3 since excessive power peaking cannot occur. ~~and the minimum required SDM is ensured.~~ The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from ~~full power conditions~~ RTP in an orderly manner and without challenging ~~plant~~ unit systems.

EDIT.

EDIT.

(continued)

BASES

ACTIONS (continued)

~~D.1.1 and D.1.2~~

~~When one or more rods are untrippable, the SDM may be adversely affected. Under these conditions, it is important to determine the SDM and, if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration to restore SDM.~~

~~In this situation, SDM verification must include the worth of the untrippable rod as well as a rod of maximum worth.~~

~~D.2~~

~~If the untrippable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours.~~

~~The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.~~

5

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

CONTROL RODS

65%

Verification that individual rods are aligned within [6.5]% of their group average height limits at a 12 hour Frequency allows the operator to detect a rod that is beginning to deviate from its expected position. ~~If the asymmetric CONTROL ROD alarm is inoperable, a Frequency of 4 hours is reasonable to prevent large deviations in CONTROL ROD alignment from occurring without detection.~~ The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

EDIT.

12

CONTROL ROD

EDIT

EDIT.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2

Verifying each CONTROL ROD is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each CONTROL ROD could result in radial tilts. Exercising each individual CONTROL ROD every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each CONTROL ROD ~~by 3%~~ enough to verify freedom of movement will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between ~~required~~ typical performances of SR 3.1.4.2 (determination of CONTROL ROD OPERABILITY by movement), if a CONTROL ROD(S) is discovered to be immovable, but is otherwise determined to be ~~trippable and aligned~~ Capable of being fully inserted, the CONTROL ROD(S) ~~is~~ may continue to be considered ~~to~~ ~~be~~ OPERABLE unless inoperable for some other reason. At any time, if a CONTROL ROD(S) is immovable, a determination of the ~~trippability (OPERABILITY) of~~ capability to fully insert (OPERABILITY) the CONTROL ROD(S) must be made, and appropriate action taken. (8) (5)

SR 3.1.4.3

<INSERT B3.1-26A>

~~Verification of rod drop time allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. The rod drop time given in the safety analysis is 1.4 seconds to 2/3 insertion. Using the identical rod drop curve gives a value of [1.66] seconds to ¾ insertion. The latter value is used in the Surveillance because the zone reference lights are located at 25% insertion intervals. The zone reference lights will activate at ¾ insertion to give an indication of the rod drop time and rod location.~~ (5)

Measuring ~~rod~~ CONTROL ROD drop times, prior to reactor criticality after reactor vessel head removal ~~and after CONTROL ROD drive system maintenance or modification~~, ensures that the reactor internals and CRDM will not interfere with CONTROL ROD motion or ~~rod~~ drop time. This Surveillance is performed during a ~~plant~~ unit outage, due to the ~~plant~~ unit conditions needed to perform the SR and the potential for an unplanned ~~plant~~ unit transient if the Surveillance were performed with the reactor at power. (9) EDIT. EDIT.

(continued)

<INSERT B 3.1-26A>

Verification of CONTROL ROD drop time allows the operator to determine that the maximum CONTROL ROD drop time permitted is consistent with the assumed CONTROL ROD drop time used in the safety analysis. The CONTROL ROD drop time given in the safety analysis is 1.66 seconds to 3/4 position insertion (Ref. 5). This 1.66 seconds includes 0.14 seconds delay time for opening of the CRD breakers and for CRDM unlatch. Using the CONTROL ROD position versus time and time versus reactivity insertion curves gives a value of 1.4 seconds to 2/3 reactivity insertion upon which the accident analysis is based (Ref. 3). The former value is used in the Surveillance because the zone reference lights are located at 25% insertion intervals. The zone reference lights will activate at 3/4 insertion to give an indication of the CONTROL ROD drop time and CONTROL ROD location. The CONTROL ROD drop time is the total elapsed time from the loss of power to the control rod drive (CRD) breaker under voltage coils until the CONTROL ROD has completed approximately 104 inches of travel from the fully withdrawn position. The safety analysis has included a CRD breaker time delay of 0.080 seconds in SAR Chapter 14 (Ref. 3). If the trip test measurement is begun with the opening of the CRD breakers, the required trip insertion time shall be reduced to 1.58 seconds and the CRD breaker time delay shall be verified to be less than or equal to 0.080 seconds.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3 (continued)

This testing is normally performed with all reactor coolant pumps operating and average moderator temperature ≥ 525°F to simulate a reactor trip under actual conditions. However, if the ~~rod~~ CONTROL ROD drop times are determined with less than four reactor coolant pumps operating, a Note allows ~~power~~ operation to continue, provided operation is restricted to the pump combination utilized during the ~~rod~~ drop time determination or pump combinations providing less total reactor coolant flow. (9)

edit

REFERENCES

1. ~~10 CFR 50, Appendix A,~~ GDC 10 and GDC 26. SAR, Section 1.4, (38)
2. 10 CFR 50.46.
3. ~~F~~SAR, Chapter ~~[14].~~ 3A and 14 EDIT.
4. ~~FSAR, Section [ ].~~ 10CFR50.36. (18)
5. ~~F~~SAR, ~~Section [ ].~~ Chapter 3 EDIT.
6. ~~FSAR, Section [ ].~~ EDIT.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Safety Rod Insertion Limit

BASES

CONTROL RODS

BACKGROUND

The insertion limits of the ~~safety and regulating rods~~ are initial condition assumptions in all safety analyses that assume ~~rod~~ insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected rod worth, and initial reactivity insertion rate. EDIT.

CONTROL ROD

The applicable criteria for the reactivity and power distribution design requirements are ~~10 CFR 50, Appendix A~~ GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). 38 edit

CONTROL ROD

in MODES 1 and 2

Limits on safety rod insertion have been established, and all ~~rod~~ positions are monitored and controlled during ~~power operation~~ to ensure that the reactivity limits, ejected rod worth, and SDM limits are preserved. EDIT.

In MODES 1 and 2, the

designated

The regulating groups are used for precise reactivity control of the reactor. The positions of the regulating groups are normally automatically controlled by the automatic control system, but they can also be manually controlled. ~~They are capable of adding negative reactivity very quickly (compared to borating).~~ ~~The~~ regulating groups must be maintained above ~~designed~~ insertion limits and are typically near the fully withdrawn position during normal operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature and fuel burnup. EDIT. EDIT.

Prior to entry into MODE 2 from MODE 3,

must be

The safety groups can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The safety groups are controlled manually by the control room operator. ~~During normal full power operation,~~ the safety groups ~~are~~ fully withdrawn. ~~The safety groups~~ must be completely withdrawn from the core prior to withdrawing any regulating groups EDIT.

(continued)

BASES

BACKGROUND (continued)

during an approach to criticality. The safety groups remain in the fully withdrawn position until the reactor is shut down. They add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

On a reactor trip, all CONTROL RODS, ~~rods (safety groups and regulating groups),~~ except the most reactive rod, are assumed to insert into the core. The safety groups shall be at their fully withdrawn limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The regulating groups may be partially inserted in the core as allowed by LCO 3.2.1, "Regulating Rod Insertion Limits." The safety group and regulating rod group insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor ~~and maintain the required SDM~~ (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from RTP ~~full power~~. The combination of regulating groups and safety groups (less the most reactive rod, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power and to achieve ~~maintain~~ the required SDM at rated no load temperature (Ref. 3). ~~The safety group insertion limit also limits the reactivity worth of an ejected safety rod.~~

EDIT. EDIT. EDIT. edit

The acceptance criteria for addressing safety and regulating rod group insertion limits and inoperability or misalignment are that:

a. There shall be no violations of:

1. specified acceptable fuel design limits, or
2. RCS pressure boundary integrity; and

b. The core must remain subcritical after an abnormality. ~~accident transients.~~ (44)

Although the SAR does not state this as an acceptance criteria for the main steam line break event, B&W has placed a design objective on this event that the core remains subcritical throughout the event (Ref. 4).

In MODES 1 and 2 while critical, the safety rod insertion limits satisfy Criteria 2 and 3 of 10CFR50.36 (Ref. 5). ~~the NRC Policy Statement.~~ (18)

In MODE 2 while subcritical, the safety rod insertion limits satisfy Criterion 4 of 10CFR50.36.

(continued)

BASES (continued)

LCO

The safety groups must be fully withdrawn any time the reactor is ~~critical or approaching criticality~~ in MODE 1 or 2. This LCO in combination with LCO 3.2.1 ensures that a sufficient amount of negative reactivity is available to shut down the reactor and ~~maintain~~ achieve the required SDM following a reactor trip. EDIT. EDIT.

APPLICABILITY

The safety groups must be within their insertion limits with the reactor in MODES 1 and 2. This LCO in combination with LCO 3.2.1 ensures that a sufficient amount of negative reactivity is available to shut down the reactor and ~~maintain~~ achieve the required SDM following a reactor trip. Refer to LCO 3.1.1 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6. EDIT.

MOVE

This LCO has been modified by a Note indicating the LCO requirement is suspended ~~during~~ for those safety rods which are inserted solely due to testing in accordance with SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the safety group to move below the LCO limits, which would normally violate the LCO. 14

ACTIONS

A.1.1, A.1.2, and A.2
~~A.1, A.2.1.1, A.2.1.2, and A.2.2~~ 40

~~When one safety rod is not fully withdrawn, 1 hour is allowed to fully withdraw the rod. This is necessary because the available SDM may be reduced with one of the safety rods not within insertion limits.~~

~~Alternatively,~~ The safety rod ~~may~~ must be declared inoperable within ~~the same~~ 1 hour time frame. This requires entry into LCO 3.1.4, "CONTROL ROD Group Alignment Limits." In addition, since the safety rod may be inserted farther than the group average insertion for a long time, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour is adequate to determine that further degradation of the SDM is not occurring.

Restoration of the required SDM, if necessary, requires increasing the boron concentration, since the ~~CONTROL ROD~~ safety rod may remain misaligned and not be providing its normal negative reactivity on tripping. ~~RCS boration must occur as described in Bases Section 3.1.1.~~ The required Completion Time of 1 hour for initiating boration is reasonable, based EDIT. 40 EDIT.

(continued)

BASES

ACTIONS A.1.1, A.1.2 and A.2 ~~A.1, A.2.1.1, A.2.1.2, and A.2.2~~ (continued) 40

on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

The allowed Completion Time of 1 hour provides an acceptable time for evaluating and repairing minor problems without allowing the unit ~~plant~~ to remain in an unacceptable condition for an extended period of time. EDIT.

B.1.1 and B.1.2 EDIT.

When more than one safety rod is not fully withdrawn, ~~inoperable~~ there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration to restore SDM.

In this situation, SDM verification must include the worth of any rod not capable of being fully inserted ~~the untrippable rod~~ as well as the CONTROL ROD ~~rod~~ of maximum worth. EDIT. 5

B.2

If more than one safety rod is not fully withdrawn ~~inoperable~~ the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from RTP ~~full power conditions~~ in an orderly manner and without challenging unit ~~plant~~ systems. EDIT. EDIT. EDIT.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that each safety rod is fully withdrawn ensures the safety rods are available to provide reactor shutdown capability. EDIT.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1 (continued)

Verification that individual safety rod positions are fully withdrawn at a 12 hour Frequency allows the operator to detect a safety rod beginning to deviate from its expected position. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of the safety rods. EDIT.

REFERENCES

1. SAR, Section 1.4, ~~10 CFR 50, Appendix A,~~ GDC 10, ~~and~~ GDC 26, and GDC 28. 38
2. 10 CFR 50.46.
3. ~~[~~SAR, ~~Section [ ]~~ Chapters 3 and 4. EDIT.
4. BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Rev. 2. 44
5. 10 CFR 50.36. 18

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

BASES

BACKGROUND

APSR

The OPERABILITY of the APSRs and ~~rod mis~~alignment are initial condition assumptions in the safety analysis that directly affect core power distributions. The applicable criteria for these power distribution design requirements are ~~10 CFR 50, Appendix A~~ GDC 10, "Reactor Design," and GDC ~~26~~, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). EDIT. 38 EDIT. 28

Mechanical or electrical failures may cause an APSR to become inoperable or to become misaligned from its group. APSR inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution. Therefore, APSR alignment and OPERABILITY are related to core operation within design power peaking limits.

Limits on APSR alignment and OPERABILITY have been established, and all ~~rod~~ positions are monitored and controlled during power operation to ensure that the power distribution limits defined by the design peaking limits are preserved. EDIT. APSR and CONTROL ROD

lc lc

~~CONTROL RODS and~~ APSRs are moved by their CONTROL ROD drive mechanisms (CRDMs). Each CRDM moves its rod ¾ inch for one revolution of the leadscrew, but at varying rates depending on the signal output ~~from the~~ Control Rod Drive Control System (CRDCS). EDIT. EDIT. EDIT.

The APSRs are arranged ~~into~~ ~~rod~~ groups that are radially symmetric. Therefore, movement of the APSRs does not introduce radial asymmetries in the core power distribution. The APSRs, which are used to assist in control of the axial power distribution, are positioned manually and do not trip. EDIT. EDIT.

LCO 3.1.6 is conservatively based on use of black (Ag-In-Cd) APSRs and bounds use of gray (Inconel) APSRs. The reactivity worth of black APSRs is greater than that of gray APSRs; thus the impact of black APSR misalignment on the core power distribution is greater.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

<Insert B3.1-34A>

~~APSR misalignment and inoperability are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing APSR inoperability or misalignment are that there shall be no violations of:~~

~~a. Specified acceptable fuel design limits; and~~

~~b. Reactor Coolant System (RCS) pressure boundary integrity.~~ (34)

~~Two types of misalignment or inoperability are distinguished.~~ (36) During movement of an APSR group, one rod may stop moving while the other rods in the group continue. This condition may cause excessive power peaking. ~~The second type of misalignment occurs when one rod drops partially or fully into the reactor core. This event causes an initial power reduction, followed by a return towards the original power, due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local linear heat rates (LHRs).~~ (36) ~~The accident analysis and reload safety evaluations define APSR insertion limits that ensure that if an APSR is stuck in or dropped in, the increase in local LHR is within the design limits. The Required Action statement in the LCO provides a conservative approach to ensure that continued operation remains within the bounds of the safety analysis (Ref. 4).~~ (34)

move

Section 3.2, "Power Distribution Limits,"

Continued operation of the reactor with a misaligned APSR is allowed if ~~AXIAL POWER IMBALANCE limits~~ are preserved. edit (34)

Because ANO-1 uses gray APSRs

The APSR alignment limits satisfy Criterion ~~2~~ 4 of ~~the NRC Policy Statement.~~ 10CFR50.36, (Ref 3). (18)

LCO

The limits on CONTROL ROD group alignment, safety rod ~~insertion~~ withdrawal, and APSR alignment, together with the limits on regulating rod insertion, APSR insertion, AXIAL POWER IMBALANCE, and QPT, ensure the reactor will operate within the fuel design criteria. The Required Action in this LCO ensures deviations from the alignment limits will be adjusted so that excessive local LHRs will not occur.

The limit for individual APSR misalignment is ~~6.5%~~ 6.5% approximately (9 inches) deviation from the group average position. This edit

(continued)

<INSERT B3.1-34A>

There are no explicit safety analyses associated with misaligned APSRs. However, alignment of the APSRs is required to prevent inducing a QUADRANT POWER TILT. The LCOs governing APSR alignment are provided because the power distribution analysis supporting LCO 3.2.1, LCO 3.2.3 and LCO 3.2.4 assumes the APSRs are aligned.

BASES

LCO (continued)

value is established based on the distance between reed switches, with additional allowances for uncertainty in the absolute position indicator amplifiers, group ~~maximum or minimum synthesizer~~, and asymmetric alarm or fault detector outputs. The position of an inoperable ~~rod~~ is not included in the calculation of the ~~rod~~ group's average position. EDIT. EDIT. EDIT.

average position calculator

APSR

Therefore, no additional uncertainties are required to be incorporated in the implementing procedures.

43

Failure to meet the requirements of this LCO may produce unacceptable ~~power peaking factors, and~~ LHRs, which may constitute initial conditions inconsistent with the safety analysis. 7

APPLICABILITY

The requirements on APSR OPERABILITY and alignment are applicable in MODES 1 and 2, ~~when the APSRs are not fully withdrawn~~ because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY and alignment of ~~rods~~ have the potential to affect the safety of the ~~plant~~. ~~OPERABILITY and alignment of the APSRs are not required when they are fully withdrawn because they do not influence core power peaking.~~ In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down ~~and not producing fission power~~ and excessive local LHRs cannot occur from APSR misalignment. 7 EDIT. EDIT. EDIT. 7 EDIT.

significant

APSRs

Unit

ACTIONS

The ACTIONS described below are required if one APSR is inoperable. The ~~plant~~ is not allowed to operate with more than one inoperable APSR. This would require the reactor to be ~~shut down~~ in accordance with LCO 3.0.3. EDIT. EDIT.

Unit

placed in MODE 3,

A.1 ~~and A.2~~ 7

An alternate to realigning a single misaligned APSR to the group average position is to align the remainder of the APSR group to the position of the misaligned or inoperable APSR, while maintaining APSR insertion, in accordance with the limits in the COLR. This restores the alignment requirements. Deviations up to 2 hours will not cause significant xenon redistribution to occur. ~~Required Action A.2~~ assumes the APSR group movement does not cause the limits of LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," to be exceeded. For this reason, 7

This alternative

(continued)

BASES

ACTIONS

A.1 ~~and A.2~~ (continued)

7

APSR group movement

~~Required Action A.1~~ is only practical for instances where small movements of the APSR group are sufficient to re-establish APSR alignment.

~~The reactor may continue in operation with the APSR misaligned if further movement of the APSR group is prohibited, so that the misalignment does not increase and cause the limits on AXIAL POWER IMBALANCE to be exceeded. The required Completion Time of up to 2 hours will not cause significant xenon redistribution to occur.~~

<INSERT B 3.1-36A>

~~B.1~~

~~The plant must be brought to a MODE in which the LCO does not apply if the Required Actions and associated Completion Times cannot be met. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging plant systems. In MODE 3, APSR group alignment limits are not required because the reactor is not generating THERMAL POWER and excessive local LHRs cannot occur from APSR misalignment.~~ 41

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification at a 12 hour Frequency that individual APSR positions are within ~~±~~6.5% of the group average height limits allows the operator to detect an APSR beginning to deviate from its expected position. ~~If the asymmetric CONTROL ROD alarm is inoperable, a 4 hour Frequency is reasonable to prevent large deviations in APSR alignment from occurring without detection.~~ 12 In addition, APSR position is continuously available to the operator in the control room so that during actual ~~rod~~ APSR motion, deviations can immediately be detected. edit

(continued)

<INSERT B 3.1-36A>

A.2

Reduction of THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER ensures that local LHR increases, due to a misaligned APSR, will not cause the core design criteria to be exceeded. The required Completion Time of 2 hours allows the operator sufficient time for reducing THERMAL POWER.

BASES (continued)

REFERENCES

1. SAR, Section 1.4, ~~10 CFR 50, Appendix A,~~ GDC 10 and GDC ~~26~~ 28. 38
2. 10 CFR 50.46.
3. ~~FSAR, Section [ ].~~
4. ~~FSAR, Section [ ].~~

3. 10CFR 50.36. 18

B 3.1 REACTIVITY CONTROL

B 3.1.7 Position Indicator Channels

BASES

the SAR discussion of | adequate | and controls are provided

BACKGROUND

within prescribed ranges, for normal operation and monitor accident conditions as appropriate to assure adequate safety

According to GDC 13 (Ref. 1), instrumentation to ~~monitor~~ maintain operating variables ~~and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE~~. LCO 3.1.7 is required to ensure OPERABILITY of the CONTROL ROD and APSR position indicators, and thereby ensure compliance with the CONTROL ROD and APSR alignment and insertion limits. 38

CONTROL RODS

The OPERABILITY, including position indication, of the ~~safety and regulating rods~~ is an initial condition assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment for the ~~safety rods, regulating rods~~ and APSRs is assumed in the safety analysis, which directly affect core power distributions and assumptions of available SDM. EDIT. EDIT.

CONTROL RODS

Local linear heat rates (LHRs)

Mechanical or electrical failures may cause a CONTROL ROD or APSR to become misaligned from its group. CONTROL ROD or APSR misalignment may cause increased ~~power peaking~~, due to the asymmetric reactivity distribution and a reduction in the total available ~~rod~~ worth for reactor shutdown. Therefore, CONTROL ROD and APSR alignment are related to core operation within design ~~power peaking~~ limits and the core design requirement of a minimum SDM. ~~Rod~~ position indication is needed to assess ~~rod~~ OPERABILITY and alignment. EDIT. EDIT. EDIT. EDIT.

CONTROL ROD

LHR

CONTROL ROD and APSR

group | and

CONTROL ROD and APSR

Limits on CONTROL ROD ~~alignment,~~ APSR alignment, and ~~safety rod~~ position have been established, and all ~~rod~~ positions are monitored and controlled during ~~power~~ operation to ensure that the power distribution and reactivity limits defined by the design ~~power peaking~~ and SDM limits are preserved. EDIT EDIT.

LHR

Three

~~Two~~ methods of CONTROL ROD and APSR position indication are provided in the CONTROL ROD Drive Control System. The ~~two~~ means are by absolute position indicator ~~and~~ relative position indicator transducers. The absolute position indicator transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the CONTROL ROD drive mechanism (CRDM) motor tube extension. three 16 EDIT.

and zone reference indicators.

(continued)

BASES

BACKGROUND (continued)

or APSR

absolute

This series of seven indicators are

Switch contacts close when a permanent magnet mounted on the upper end of the CONTROL ROD assembly (CRA) leadscrew extension comes near. As the leadscrew and CRA move, the switches operate sequentially, producing an analog voltage proportional to position. Other reed switches included in the same tube with the position indicator matrix provide full in and full out limit indications, and absolute position indications at 0%, 25%, 50%, 75%, and 100% travel. (called zone reference indicators). The relative position indicator transducer is a potentiometer, driven by a step motor that produces a signal proportional to CONTROL ROD or APSR position, based on the electrical pulse steps that drive the CRDM.

CONTROL ROD or APSR edit

edit

edit

edit

edit

edit

edit

~~Two absolute position indicator channel designs may be used in the unit: type A absolute position indicators and type A-R4C absolute position indicators. The type A absolute position indicator transducer is a voltage divider circuit made up of 48 resistors of equal value connected in series. One end of 48 reed switches is connected at a junction between each of the resistors, so that as the magnet mounted on the leadscrew moves, either one or two reed switches are closed in the vicinity of the magnet. The type A-R4C (redundant four channel) absolute position indicator transducer has two parallel sets of voltage divider circuits made up of 36 resistors each, connected in series (channels A and B). One end of 36 reed switches is connected at a junction between each of the resistors of the two parallel circuits. The reed switches making up each circuit are offset, such that the switches for channel A are staggered with the switches for channel B. The type A-R4C is designed such that either two or three reed switches are closed in the vicinity of the magnet. By its design, the type A-R4C absolute position indicator provides redundancy, with the two three sequence of pickup and drop out of reed switches to enable a continuity of position signal when a single reed switch fails to close.~~

edit

and APSR

rod

individual position indication

CONTROL ROD or APSR

CONTROL ROD position indicating readout devices located in the control room consist of single ~~CRA~~ position meters on a ~~wall mounted~~ position indication panel and ~~four~~ group average position meters ~~on the console~~. A selector switch permits either relative or absolute position indication to be displayed on all of the ~~single rod~~ meters. Indicator lights are provided on the ~~single CRA meter~~ panel to indicate when each ~~CRA~~ is fully withdrawn, fully inserted,

edit

edit

edit

edit

edit

edit

edit

(continued)

BASES

EDIT.

BACKGROUND (continued)

enabled, or transferred, and whether a ~~ERA~~ rc rod position asymmetry alarm condition is present. Additional Indicators ~~on the console~~ show full insertion, full withdrawal, and enabled for motion for each CONTROL ROD group. EDIT. ~~Identical instrumentation and devices exist for the APSR group.~~ EDIT. The consequence of continued operation with an inoperable absolute position indicator or relative position indicator channel is a decreased reliability in determining CONTROL ROD and APSR position. EDIT. Therefore, the potential for operation in violation of design LHR ~~peaking factors~~ or SDM Limits is increased. EDIT.

APPLICABLE SAFETY ANALYSES

CONTROL ROD and APSR position accuracy is essential during power operation. EDIT. LHR ~~Power peaking~~, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2) with CONTROL RODS or APSRs operating outside their limits undetected. EDIT. ~~Regulating rod, safety rod~~ CONTROL ROD and APSR positions must be known in order to verify the core is operating within the group sequence, overlap, design LHRs ~~peaking limits~~, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Safety Rod Insertion Limits"; LCO 3.2.1, "Regulating Rod Insertion Limits"; and LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"). EDIT. The ~~rod~~ CONTROL ROD and APSR positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "CONTROL ROD Group Alignment Limits," and LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits"). EDIT. CONTROL ROD and APSR positions are continuously monitored to provide operators with information that ensures the ~~plant~~ unit is operating within the bounds of the accident analysis assumptions.

¶ In MODES 1 and 2 while critical, the ~~The~~ CONTROL ROD and APSR position indicator channels satisfy Criterion 2 of 10CFR50.36 (Ref. 3). ~~of the NRC Policy Statement. The CONTROL ROD position indicators monitor CONTROL ROD position, which is an accident initial condition.~~ 18

In MODE 2 while subcritical, the CONTROL ROD and APSR position indicator channels satisfy Criterion 4 of 10CFR50.36. 16

LCO

LCO 3.1.7 specifies that one absolute position indicator channel ~~and one relative position indicator channel~~ be OPERABLE for each CONTROL ROD and APSR.

~~The agreement between the relative position indicator channel and the absolute position indicator channel, within the limit given in the COLR, indicates that relative position indicators are adequately calibrated and can be~~ 16

(continued)

BASES

LCO (continued)

~~used for indication of the measurement of CONTROL ROD group position. A deviation of less than the allowable limit, given in the COLR, in position indication for a single CONTROL ROD or APSR, ensures confidence that the position uncertainty of the corresponding CONTROL ROD group or APSR group is within the assumed values used in the analysis that specifies CONTROL ROD group and APSR insertion limits.~~ (16)

This ~~These~~ requirement~~s~~ ensures that CONTROL ROD and APSR position indication during MODES 1 and 2 ~~power operation~~ and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channel~~s~~ ensures that inoperable, misaligned, or mispositioned CONTROL RODS or APSRs can be detected. Therefore, LHR ~~power peaking~~ and SDM can be controlled within acceptable limits. EDIT. EDIT. (16) EDIT

APPLICABILITY

In MODES 1 and 2, OPERABILITY of the position indicator channel~~s~~ is required, since the reactor is, or is capable of, generating THERMAL POWER in these MODES. In MODES 3, 4, 5, and 6, Applicability is not required because the reactor is shut down with the required minimum SDM and is not generating significant THERMAL POWER. (16) EDIT.

ACTIONS

~~A.1~~

~~If the relative position indicator channel is inoperable for one or more rods, the position of the rod(s) is still monitored by the absolute position indicator channel for each affected rod. The absolute position indicator channel may be used if it is determined to be OPERABLE. The required Completion Time of 8 hours is reasonable to provide adequate time for the operator to determine position indicator channel status. Continuing the verification every 8 hours thereafter in the applicable condition is acceptable, based on the fact that during normal power operation excessive movement of the groups is not required. Also, if the rod is out of position during this 8 hour period, the simultaneous occurrence of an event sensitive to the rod position has a small probability.~~ (16)

(continued)

BASES

ACTIONS (continued)

B.1.1

If the absolute position indicator channel is inoperable for one or more rods, the position of the rod(s) is monitored by the relative position indicator channel for each affected rod. However, the relative position indicator channel is not as reliable a method of monitoring rod position as the absolute position indicator because it counts electrical pulse steps driving the CRDM motor rather than actuating a switch located at a known elevation. Therefore, the affected rod's position can be determined with more certainty by actuating one of its zone reference indicator switches located at discrete elevations. The required Completion Time of 8 hours provides the operator adequate time for adjusting the affected rod's position to an appropriate zone reference indicator location. If the rod is out of position during this 8 hour period, the simultaneous occurrence of an event sensitive to the rod position has a small probability.

B.1.2 16

To allow continued operation, the rods with inoperable absolute position indicator channels are maintained at the zone reference indicator position. In addition, the affected rods are maintained within the limits of LCO 3.1.5 (when the affected rod is a safety rod); LCO 3.2.1 (when the affected rod is a regulating rod); or LCO 3.2.2 (when the affected rod is an APSR). This Required Action ensures safety rods remain fully withdrawn, and that regulating rods and APSRs remain aligned within their insertion limits. The required Completion Time of 8 hours is reasonable for allowing the operator adequate time to determine the affected rods are in compliance with these LCOs. Continuing to verify the rod positions every 8 hours thereafter is reasonable for ensuring that rod alignment and insertion are not changing, and provides the operator adequate time to correct any deviation that may occur. Continuing the verification every 8 hours thereafter in the applicable condition is acceptable, based on the fact that during normal power operation excessive movement of the groups is not required. Also, if the rod is out of position during this 8 hour period, the simultaneous occurrence of an event sensitive to the rod position has a small probability.

(continued)

BASES

ACTIONS (continued)

B.2.1

If the absolute position indicator is inoperable for one or more rods, the position of the rod is monitored by the relative position indicator channel for each affected rod. However, the relative position indicator channel is not as reliable a method of monitoring rod position as the absolute position indicator because it counts electrical pulse steps. The fixed incore system can be used to indirectly determine the absolute position of the affected rod. The fixed incore instrumentation can provide a continual update of CONTROL ROD position, therefore this method can be used to allow continued operation of the reactor with a manual CONTROL ROD movement, while maintaining verification of CONTROL ROD insertion and alignment. Required Action B.2.1. restricts rod motion by placing the groups with nonindicating rods in manual control; thus, even if the rod fails to move in alignment with the group, misalignment is limited. The required Completion Time of 8 hours provides the operator adequate time for placing the rods in manual control, and is consistent with the required Completion Time for Required Action B.1.1. If the rod is out of position during this 8 hour period, the simultaneous occurrence of an event sensitive to the rod position has a small probability.

16

B.2.2

Continuing to verify the rod positions every 8 hours is reasonable for ensuring that rod alignment and insertion are not changing, and provides the operator adequate time to correct any deviation that may occur. The additional Completion Time of 1 hour after motion of nonindicating rods, which exceeds 15 inches in one direction since the last determination of the rod's position, ensures that the rod with inoperable position indication will not be misaligned for a significant period of time, in the event the rod is moved. The specified Completion Times are acceptable because the simultaneous occurrence of a mispositioned rod and an event sensitive to the rod position has a small probability.

(continued)

BASES

ACTIONS (continued)

A.1
~~C.1~~

required

If ~~both~~ the ~~absolute~~ position indicator channel ~~and relative position indicator channel are~~ is inoperable for one or more rods, ~~or if the Required Actions and associated Completion Times are not met,~~ the position of the CONTROL ROD or APSR ~~rod(s)~~ is not known with certainty. Therefore, each affected ~~rod~~ CONTROL ROD or APSR must be declared inoperable, and the limits of LCO 3.1.4 or LCO 3.1.6 apply. The required Completion Time for declaring the rod(s) inoperable is immediately. Therefore LCO 3.1.4 or LCO 3.1.6 is entered immediately, and the required Completion Times for the appropriate Required Actions in those LCOs apply without delay.

16 edit. edit.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1 <INSERT B 3.1-44 A> 16

~~Verification is required that the Absolute Position Indicator channels and Relative Position Indicator channels agree within the limit given in the COLR. This verification ensures that the Relative Position Indicator channels, which are regarded as the potentially less reliable means of position indication, remain OPERABLE and accurate.~~ 16 The required Frequency of 12 hours is adequate for verifying that no degradation in system OPERABILITY has occurred. ~~If the asymmetric CONTROL ROD alarm is inoperable, then the Surveillance is performed every 4 hours. This required Frequency is adequate for ensuring that the CONTROL RODS and APSRs do not exceed their alignment limits.~~ 12

<INSERT B3.1-44 B> 16

REFERENCES

1. SAR, Section 1.4, ~~10 CFR 50, Appendix A,~~ GDC 13. 38 EDIT.

2. ~~FSAR, Section [14.1.2.2], Section [14.1.2.3], Section [14.1.2.6], Section [14.1.2.7], Section [14.2.2.4], and Section [14.2.2.5].~~ Chapter 14.

3. 10 CFR 50.36. 18

<INSERT B3.1-44A>

A CHANNEL CHECK of the required position indication channel ensures that position indication for each CONTROL ROD and APSR remains OPERABLE and accurate. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. However, this CHANNEL CHECK will be used to detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

When compared to other channels, the agreement criteria between the channels is determined by the unit staff. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The CHANNEL CHECK supplements less formal but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required position indicator channel.

<INSERT B3.1-44B>

SR 3.1.7.2

A CHANNEL CALIBRATION of the required position indication channel verifies that the channel responds within the necessary range and accuracy.

The Frequency of 18 months is based on operating experience and consistency with the typical industry refueling cycle.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions Systems—MODE 1

BASES

BACKGROUND

The purpose of this ~~MODE 1~~ LCO is to permit PHYSICS TESTS to be conducted by providing exemptions from the requirements of other LCOs. Establishment of a test program to verify that structures, systems, and components will perform satisfactorily in service is required by Section XI of 10 CFR 50, Appendix B (Ref. 1). Testing is required as an integral part of the design, fabrication, construction, and operation of the unit ~~power plant~~. All functions necessary to ensure that specified design conditions are not violated during normal operation and abnormalities ~~anticipated operational occurrences~~ must be tested. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2). EDIT. EDIT. EDIT.

The key objectives of a test program are to ~~(Ref. 3)~~: 15

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality; during startup, low power operations, and power ascension; at high powers; and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. ~~4~~ 3). EDIT. 15

<INSERT B 3.1-45A>

PHYSICS TESTS procedures are written and approved in accordance with established guidelines. The procedures include all information necessary to permit a detailed

(continued)

<INSERT B 3.1-45A>

The inclusion of this PHYSICS TESTS Exception LCO is acceptable based on the use of approved written procedures, administrative controls, the requirements of 10CFR50.59, and the LCO 3.1.8 provisions in effect during the conduct of PHYSICS TESTS.

BASES

BACKGROUND (continued)

execution of testing required to ensure the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation. Examples of PHYSICS TESTS include determination of critical boron concentration, CONTROL ROD group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because reactor protection criteria are preserved by the LCOs still in effect and by the SRs. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on ~~nuclear hot channel factors,~~ linear heat rate (LHR), ejected rod worth, and shutdown capability are maintained during the PHYSICS TESTS. 20 EDIT.

Reference ~~8~~ 4 ~~defines requirements for~~ describes the initial testing of the facility, including PHYSICS TESTS. Tables ~~13.2 and 13.3~~ 13-2 (Ref. ~~6~~ 5) summarizes the ~~zero, low power, and power~~ post-criticality tests. Requirements for reload fuel cycle PHYSICS TESTS are given in ~~Table 1 ANSI/ANS-19.6.1-1985~~ SAR Section 3A.9 (Ref. ~~6~~ 3). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, one or more LCOs must sometimes be suspended to make completion of PHYSICS TESTS possible or practical. EDIT. EDIT. 15

This is acceptable as long as the fuel design criteria are not violated. When one or more of the limits specified in:

LCO 3.1.4, "CONTROL ROD Group Alignment Limits";
LCO 3.1.5, "Safety Rod Insertion Limits";
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits";
LCO 3.2.1, "Regulating Rod Insertion Limits," for the restricted operation region only;
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits";
LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits"; ~~or~~ and EDIT.
LCO 3.2.4, "QUADRANT POWER TILT (QPT)" 15

are suspended for PHYSICS TESTS, the fuel design criteria are preserved by maintaining the ~~nuclear hot channel factors~~ LHR (in MODE 1 PHYSICS TESTS) within ~~their~~ limits, maintaining ejected rod worth within limits by restricting regulating rod insertion to within the acceptable operating region or the restricted operating region, by limiting maximum THERMAL POWER and by maintaining SDM ~~≥ 1.0% Δk/k~~ within the limit provided in the COLR. Therefore, 20 1

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

LHR

surveillance of the $F_Q(Z)$, the $F_{\Delta H}$ and SDM is required to verify that their limits are not exceeded. The limits for the ~~nuclear hot channel factors~~ are specified in the COLR. Refer to the Bases for LCO 3.2.5 for a complete discussion of $F_Q(Z)$ and $F_{\Delta H}$. During PHYSICS TESTS, one or more of the LCOs that normally preserve the $F_Q(Z)$ and $F_{\Delta H}$ limits may be suspended. However, the results of the safety analysis are not adversely impacted if verification that $F_Q(Z)$ and $F_{\Delta H}$ are within their limits is obtained, while one or more of the LCOs is suspended. Therefore, SRs are placed on $F_Q(Z)$ and $F_{\Delta H}$ during MODE 1 PHYSICS TESTS to verify that these factors remain within their limits. Periodic verification of these factors allows PHYSICS TESTS to be conducted while continuing to maintain the design criteria. (20)

LHR

Core LHRs

LHR

Core LHRs

when THERMAL POWER exceeds 20% RTP

PHYSICS TESTS include measurement of core nuclear parameters or exercise of control components that affect process variables. Among the process variables involved are AXIAL POWER IMBALANCE and QPT, which represent initial condition input (power peaking) for the accident analysis. Also involved are the movable control components, i.e., the regulating rods and the APSRs, which affect power peaking. ~~and are required for shutdown of the reactor.~~ The limits for these variables are specified for each fuel cycle in the COLR. EDIT.

~~PHYSICS TESTS satisfy Criteria 1, 2, and 3 of the NRC Policy Statement.~~ (13)

<INSERT B3.1-47A>

LCO

and APSRs (15)

This LCO permits individual CONTROL RODS to be positioned outside of their specified group alignment and withdrawal limits and to be assigned to other than specified CONTROL ROD groups, and permits AXIAL POWER IMBALANCE and QPT limits to be exceeded during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics and nuclear instrumentation operation.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1 (for the restricted operation region only), LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4 may be suspended during the performance of PHYSICS TESTS provided: (15)

a. THERMAL POWER is maintained ≤ 85% RTP;

(continued)

<INSERT B 3.1-47A>

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10CFR50.36 (Ref. 6) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria for the other LCOs is provided in their respective Bases.

BASES

LCO (continued)

b. Nuclear overpower trip setpoint is ≤ 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP;

20

c. LHR is ~~$F_Q(Z)$ and $F^N_{\Delta H}$ are~~ maintained within limits specified in the COLR while operating at greater than 20% RTP; and

1

d. SDM is ~~maintained ≥ 1.0% Δk/k~~ verified to be within the limit provided in the COLR

Operation with THERMAL POWER ≤ 85% RTP during PHYSICS TESTS provides an acceptable thermal margin when one or more of the applicable LCOs is out of specification. Eighty-five percent RTP is consistent with the maximum power level for conducting the intermediate core power distribution test specified in Reference 3 ~~4~~. The nuclear overpower trip setpoint is reduced so that a similar margin exists between the steady state condition and trip setpoint as exists during normal operation at RTP. edit

<INSERT B3.1-48A> 20

APPLICABILITY

This LCO is applicable in MODE 1, when the reactor has completed low power testing and is in power ascension, or during power operation with THERMAL POWER > 5% RTP but ≤ 85% RTP. This LCO is applicable for power ascension testing, as ~~defined by Regulatory Guide 1.68~~ described in SAR Section 3A.9 (Ref. 3). In MODE 2, Applicability of this LCO is not required because LCO 3.1.9, "PHYSICS TESTS Exceptions—MODE 2," addresses PHYSICS TESTS exceptions initiated in MODE 2. In MODES 3, 4, 5, and 6, Applicability is not required because PHYSICS TESTS are not performed in these MODES. 15 24

ACTIONS

A.1 and A.2

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the ~~plant~~ unit conditions. Boration will be continued until SDM is within limit. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. EDIT

(continued)

<INSERT B 3.1-48A>

LCO provision c is modified by a Note that requires the adherence to LHR requirements only when THERMAL POWER is greater than 20% RTP. This establishes an LCO provision that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

BASES

ACTIONS

A.1 and A.2 (continued)

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification. A Completion Time of one hour is provided for the operator to restore Compliance with the excepted LCOs.

EDIT.

B.1

If THERMAL POWER exceeds 85% RTP, then 1 hour is allowed for the operator to reduce THERMAL POWER to within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCO, addressed by PHYSICS TESTS exceptions.

If the nuclear overpower trip setpoint is not within the specified limits, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCO, addressed by these PHYSICS TESTS exceptions.

If the results of the incore flux map indicate that either $F_Q(Z)$ or $F_{\Delta H}^N$ LHR has exceeded its limit, then PHYSICS TESTS are suspended. This action is required because of direct indication that the core peaking factors LHR, which are is a fundamental initial conditions for the safety analysis, are is excessive. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification. (20)

<INSERT B3.1-49A>

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that THERMAL POWER is ≤ 85% RTP ensures that the required additional thermal margin has been established prior to and during PHYSICS TESTS. The required Frequency of once per hour allows the operator adequate time to

(continued)

<INSERT B 3.1-49A>

This Condition is modified by a Note that requires performance of the Required Action only when THERMAL POWER is greater than 20% RTP. This establishes an ACTIONS entry Condition that is consistent with LCO provision c and the Applicability of LCO 3.2.5, "Power Peaking."

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1 (continued)

determine any degradation of the established thermal margin during PHYSICS TESTS.

SR 3.1.8.2

Verification that Core LHRs ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ are within their limits ensures that core LHR ~~local linear heat rate~~ and departure from nucleate boiling ratio will remain within their limits, while one or more of the LCOs that normally control these design limits are out of specification. The required Frequency of 2 hours allows the operator adequate time for collecting a flux map and for performing the LHR ~~hot channel factor~~ verification~~s~~, based on operating experience. If SR 3.2.5.1 is not met, PHYSICS TESTS are suspended and LCO 3.2.5 applies. This Frequency is more conservative than the Completion Time for restoration of the individual LCOs that preserve the LHR ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ limits. (20)

<INSERT B3.1-50A> →

SR 3.1.8.3

Verification that the nuclear overpower trip setpoint is within the limit specified for each PHYSICS TEST ensures that core protection at the reduced power level is established ~~and will remain in place~~ during the PHYSICS TESTS. Performing the verification once prior to the performance of PHYSICS TESTS ~~every 8 hours~~ allows the operator adequate time for verifying ~~determining any degradation of~~ the established trip setpoint ~~margin~~ before initiating ~~and during~~ PHYSICS TESTS ~~, and for adjusting the nuclear overpower trip setpoint~~. (2)

SR 3.1.8.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. Reactor Coolant System (RCS) boron concentration;

b. CONTROL ROD position;

c. Doppler defect; ~~RCS average temperature;~~ (39)

(continued)

<INSERT B 3.1-50A>

This SR is modified by a Note that requires performance only when THERMAL POWER is greater than 20% RTP. This establishes a performance requirement that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.4 (continued) EDIT.

d. Fuel burnup based on gross thermal energy generation;

e. Samarium concentration;

f. ~~e.~~ ~~Xenon concentration; and~~

Moderator defect

g. ~~f.~~ ~~Isothermal temperature coefficient (ITC)~~

~~Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the RCS.~~ 39

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59, EDIT.

~~3. Regulatory Guide 1.68, Revision 2, August 1978.~~ 15

~~4. ANSI/ANS-19.6.1-1985, December 13, 1985.~~

4. ~~5.~~ ~~F~~SAR, Section ~~[13.4.8]~~ 13.3, 13.4 and 13.6. EDIT.

5. ~~6.~~ ~~F~~SAR, Section 13.4, Table 13-2. ~~[13.4.8], [Tables 13-2 and 13-4, Am. 49, September 30, 1976]~~ EDIT.

6. 10 CFR 50.36. 18

3. SAR, Section 3A.9 15

B 3.1 REACTIVITY CONTROL SYSTEMS

edit

B 3.1.9 PHYSICS TESTS Exceptions—MODE 2

BASES

BACKGROUND

The purpose of this MODE 2 LCO is to permit PHYSICS TESTS to be conducted by providing exemptions from the requirements of other LCOs. Establishment of a test program to verify that structures, systems, and components will perform satisfactorily in service is required by Section XI of 10 CFR 50, Appendix B (Ref. 1). Testing is required as an integral part of the design, fabrication, construction, and operation of the unit ~~power plant~~. All functions necessary to ensure that specified design conditions are not violated during normal operation and abnormalities ~~anticipated operational occurrences~~ must be tested. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

edit

edit

edit

The key objectives of a test program are to ~~(Ref. 3)~~:

edit

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality; during startup, low power operations, and power ascension; at high powers; and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the core are consistent with the design predictions, and that the core can be operated as designed (Ref. ~~4~~ 3).

<Insert B 3.1-52A>

PHYSICS TESTS procedures are written and approved in accordance with established guidelines. The procedures include all information necessary to permit a detailed

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(continued)

<INSERT B 3.1-52A>

The inclusion of this PHYSICS TESTS Exception LCO is acceptable based on the use of approved written procedures, administrative controls, the requirements of 10CFR50.59, and the LCO 3.1.9 provisions in effect during the conduct of PHYSICS TESTS.

BASES

BACKGROUND (continued)

execution of testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation.

Examples of MODE 2 PHYSICS TESTS include determination of critical boron concentration, CONTROL ROD group worth, and reactivity coefficients.

APPLICABLE SAFETY ANALYSES

Reference ~~8~~ 4 ~~defines requirements for~~ describes the initial testing of the facility, including PHYSICS TESTS. Tables ~~13-3 and 13-4~~ 13-2 (Ref. ~~6~~ 5) summarizes the ~~zero, low power, and power~~ post-criticality tests. Requirements for reload fuel cycle PHYSICS TESTS are given in ~~Table 1 of ANSI/ANS-19.6.1-1985~~ SAR Section 3A.9 (Ref. ~~4~~ 3). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more of the LCOs must be suspended to make completion of PHYSICS TESTS possible or practical.

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It is acceptable to suspend the following LCOs for PHYSICS TESTS because reactor protection criteria are preserved by the LCOs still maintained and by the SRs:

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.4, "CONTROL ROD Group Alignment Limits";
LCO 3.1.5, "Safety Rod Insertion Limits";
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits";
LCO 3.2.1, "Regulating Rod Insertion Limits" ~~for the restricted operation region only; and~~ 42
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality." 15

Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on THERMAL POWER and shutdown capability are maintained during the PHYSICS TESTS.

Shutdown capability is preserved by limiting ~~maximum obtainable~~ THERMAL POWER and maintaining adequate SDM, when in MODE 2 PHYSICS TESTS. In MODE 2, the Reactor Coolant System (RCS) temperature must be within the narrow range instrumentation for ~~plant~~ Unit control. The narrow range temperature instrumentation goes on scale at 520°F. Therefore, it is considered safe to allow the minimum RCS

EDIT.
EDIT.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

temperature to decrease to 520°F during MODE 2 PHYSICS TESTS, based on the low probability of an accident occurring and on prior operating experience.

PHYSICS TESTS include measurement of core nuclear parameters or exercise of control components that affect process variables.

~~PHYSICS TESTS satisfy Criteria 1, 2, and 3 of the NRC Policy Statement.~~ 13

<INSERT B 3.1-54B>

LCO

This LCO permits individual CONTROL RODS and APSRs to be positioned outside of their specified group alignment and withdrawal limits and to be assigned to other than specified CONTROL ROD groups during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics. ED IT.

This LCO also allows suspension of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, ~~and LCO 3.4.2~~, provided: 15

a. THERMAL POWER is ≤ 5% RTP;

b. Nuclear overpower trip setpoints on the OPERABLE nuclear power range channels are set to ≤ ~~25%~~ 5% RTP; 23

c. Nuclear instrumentation ~~source range and intermediate range~~ high startup rate CONTROL ROD withdrawal inhibit ~~are~~ is OPERABLE; and 25 15

d. SDM is maintained ~~≥ 1.0% Δk/k~~ within the limit provided in the COLR. 1 EDIT.

The limits of LCO 3.2.3 and LCO 3.2.4 ~~do not apply in MODE 2~~ are not exempted by this specification because they. Inhibiting CONTROL ROD withdrawal, based on startup rate, also limits local linear heat rate (LHR), departure from nucleate boiling ratio (DNBR), and peak RCS pressure during accidents initiated from low power.

APPLICABILITY

This LCO is applicable ~~in MODE 2~~ when the reactor is either subcritical or ~~not~~ critical ~~or when~~ with THERMAL POWER ~~is~~ ≤ 5% RTP. This LCO is applicable for initial criticality or low power testing, as ~~defined by Regulatory Guide 1.68~~ described in SAR, Section 3A.9 (Ref. 3). In MODE 1, 24 15

<INSERT B 3.1-54A>

(continued)

<INSERT B 3.1-54A>

The Applicability is stated as "during PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.

<INSERT B 3.1-54B>

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10CFR50.36 (Ref. 6) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria for the other LCOs is provided in their respective Bases.

BASES

APPLICABILITY (continued)

Applicability of this LCO is not required because LCO 3.1.8, "PHYSICS TESTS Exceptions," addresses PHYSICS TESTS exceptions in MODE 1. In MODES 3, 4, 5, and 6, ~~Applicability~~ a test exception LCO is not required because ~~physics testing is not performed in these MODES~~ the excepted LCOs do not apply in these MODES. EDIT. EDIT.

ACTIONS

A.1

If THERMAL POWER exceeds 5% RTP, a positive reactivity addition could be occurring, and a nuclear excursion could result. To ensure that local LHR, DNBR, and RCS pressure limits are not violated, the reactor is immediately tripped. The necessary prompt action requires manual operator action to open the CONTROL ROD drive trip breakers without attempts to reduce THERMAL POWER by actuating the control system (i.e., CONTROL ROD insertion or RCS boration). EDIT. EDIT.

B.1 and B.2

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the ~~plant~~ unit conditions. Boration will be continued until SDM is within limit. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. EDIT.

Suspension of PHYSICS TESTS exceptions requires restoration ~~of each of the applicable LCOs to within specification.~~ A Completion Time of one hour is provided for the operator to restore compliance with the excepted LCOs. EDIT.

C.1

If the nuclear overpower trip setpoint is > ~~25%~~ 5% RTP, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification, in order to ensure that continuity of reactor operation is within initial condition limits. This required Completion 23

(continued)

BASES

ACTIONS

C.1 (continued)

Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

If the nuclear instrumentation ~~source and intermediate range~~ high startup rate CONTROL ROD withdrawal inhibit function~~s~~ ~~are~~ inoperable, then 1 hour is allowed for the operator to restore the functions to OPERABLE status or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions. (25) (15)

<INSERT B3.1-56 A> (25)

SURVEILLANCE REQUIREMENTS

~~SR 3.1.9.1~~

~~Performing a CHANNEL FUNCTIONAL TEST on each nuclear instrumentation source and intermediate range high startup rate CONTROL ROD withdrawal inhibit and nuclear overpower channel, ensures that the instrumentation required to detect a deviation from THERMAL POWER or to detect a high startup rate is OPERABLE. Performing the test once within 24 hours, prior to initiating PHYSICS TESTS, ensures that the instrumentation is OPERABLE shortly before PHYSICS TESTS begin and allows the operator to correct any instrumentation problems.~~ (26)

SR 3.1.9.~~2~~ 1 (26)

Verification that THERMAL POWER is ≤ 5% RTP ensures that ~~an~~ ~~adequate margin is maintained between the THERMAL POWER level and the nuclear overpower trip setpoint.~~ Hourly verification is adequate for the operator to determine any change in core conditions, such as xenon redistribution occurring after a THERMAL POWER reduction, that could cause THERMAL POWER to exceed the specified limit. (23)

local LHR, DNBR, and RCS pressure limits are not violated and that entry into Actions Condition A is performed promptly.

(continued)

<INSERT B 3.1-56A>

The nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit function is not required when the reactor power level is above the operating range of the instrumentation channel. For example, if the reactor power level is above the source range channel operating range, then only the intermediate range high startup rate CONTROL ROD withdrawal inhibit is required to be functional.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.9.~~3~~ 2 — 26

Verification that the nuclear overpower trip setpoint is within the limit specified for PHYSICS TESTS ensures that core protection at the reduced power level is established ~~and will remain in place~~ during PHYSICS TESTS. Performing the verification ~~once per 8 hours~~ prior to the performance of PHYSICS TESTS allows the operator adequate time for ~~determining any degradation of~~ verifying the established trip setpoint ~~margin~~ before ~~and during~~ initiating PHYSICS TESTS~~, and for adjusting the nuclear overpower trip setpoint~~. — 2

SR 3.1.9.~~4~~ 3 — 26

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects: — 39

a. RCS boron concentration;

b. CONTROL ROD position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Samarium concentration;

~~e.~~ f. Xenon concentration; ~~and~~

~~f.~~ g. Isothermal temperature coefficient (ITC), when below the point of adding heat (POAH);

h. Moderator defect, when above the POAH; and

i. Doppler defect, when above the POAH.

Using the ITC accounts for Doppler reactivity in this calculation ~~because~~ when the reactor is subcritical, or critical but below the POAH, and the fuel temperature will be changing at the same rate as the RCS. — 39

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. ~~Regulatory Guide 1.68, Revision 2, August 1978.~~ — 15
3. SAR, Section 3A.9.

(continued)

BASES

REFERENCES (continued)

~~4. ANSI/ANS-19.6.1-1985, December 13, 1985~~ 15

4. ~~5.~~ ~~F~~SAR, Section ~~[13.4.8]~~ 13.3, 13.4 and 13.6. EDIT.

5. ~~6.~~ ~~F~~SAR, Section ~~[13.4.8], [Table 13-3 and Table 13-4]~~ 13.4, Table 13-2. EDIT.

6. 10CFR 50.36. 18