Duke Power
A Duke Energy Company

OCONEE NUCLEAR STATION
IMPROVED TECHNICAL SPECIFICATIONS

VOLUME 1

SPLIT REPORT
AND
SECTIONS
1.0 2.0 3.0

DUKE POWER
Generating Excellence
Oconee Nuclear Site

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APPLICATION OF SCREENING CRITERIA

TO THE

OCONEE
UNITS 1, 2 AND 3
TECHNICAL SPECIFICATIONS

CONTENTS

1. INTRODUCTION

The purpose of this document is to confirm the results of the Babcock and Wilcox Owners Group application of the Technical Specification screening criteria on a plant specific basis for the Oconee Nuclear Station (ONS) Units 1, 2 and 3. Duke Energy has reviewed the application and confirmed the applicability of the screening criteria to each of the Technical Specifications utilized in the following documents: 1) BAW-1923, Volume I, "Justification and Background for Technical Specification Improvements" submitted by letter dated February 16, 1987; 2) 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; 3) NRC Staff Review of Nuclear Steam Supply Vendor Owners Groups Application of the Commissions Interim Policy Statement Criteria to Standard Technical Specifications (Wilgus/Murley letter dated May 9, 1988); and 4) NUREG-1430, "Standard Technical Specifications, Babcock & Wilcox Plants" (Reference 2) and applied the criteria to each of the current ONS Units 1, 2 and 3 Technical Specifications. Additionally, in accordance with the NRC Final Policy Statement (Reference 3) and 10 CFR 50.36, this confirmation of the application of screening criteria includes confirming the risk insights from Probabilistic Risk Assessment (PRA) evaluations, provided in Reference 1, as applicable to the ONS Units 1, 2 and 3.

2. SCREENING CRITERIA

Duke Energy has utilized the screening criteria provided in 10 CFR 50.36 to develop the results contained in the attached matrix. PRA insights as used in the Babcock and Wilcox Owners Group submittal were utilized, confirmed by Duke Energy, and are discussed in the next section of this report. The screening criteria and discussion provided in Reference 3 are as follows:

> <u>Criterion 1:</u> Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary:
>
> <u>Discussion of Criterion 1:</u> A basic concept in the adequate protection of the public health and safety is the prevention of accidents. Instrumentation is installed to detect significant abnormal degradation of the reactor coolant pressure boundary so as to allow operator actions to either correct the condition or to shut down the plant safely, thus reducing the likelihood of a loss-of-coolant accident.
>
> This criterion is intended to ensure that Technical Specifications control those instruments specifically installed to detect excessive reactor coolant system leakage. This criterion should not, however, be interpreted to include instrumentation to detect precursors to reactor coolant pressure boundary leakage or instrumentation to identify the source of actual leakage (e.g., loose parts monitor, seismic instrumentation, valve position indicators).
>
> <u>Criterion 2:</u> A process variable, design feature, or operating restriction that is an initial condition of a design basis accident (DBA) or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier:
>
> <u>Discussion of Criterion 2:</u> Another basic concept in the adequate protection of the public health and safety is that the plant shall be operated within the bounds of the initial conditions assumed in the existing design basis accident and transient analyses and that the plant will be operated to preclude unanalyzed transients and accidents. These analyses consist of postulated events, analyzed in the Updated Final Safety Analysis Report (UFSAR), for which a structure, system, or component must meet specified functional goals. These analyses are contained in Chapters 6 and 15 of the UFSAR (or equivalent chapters) and are identified as Condition II, III, or IV events (ANSI N18.2) (or equivalent) that either assume the failure of or present a challenge to the integrity of a fission product barrier.
>
> 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

2. (continued)

and controlled from the control room. These could also include other features or characteristics that are specifically assumed in 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 acceptably low. This criterion also includes active design features (e.g., high pressure/low pressure system valves and interlocks) and operating restrictions (pressure/temperature limits) needed to preclude unanalyzed accidents and transients.

<u>Criterion 3:</u> 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:

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

A safety sequence analysis is a systematic examination of the actions required to mitigate the consequences of events considered in the plant's design basis accident and transient analyses, as presented in Chapters 6 and 15 of the plant's Final Safety Analysis Report (or equivalent chapters). Such a safety sequence analysis considers all applicable events, whether explicitly or implicitly presented. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria), so that the plant response to design basis accidents and transients limits the consequences of these events to within the appropriate acceptance criteria.

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).

2. (continued)

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

<u>Discussion of Criterion 4</u>: It is the Commission 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, or components that meet this criterion:

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

The Commission recognizes that other structures, systems, or components may meet this criterion. Plant and design-specific PSA's have yielded valuable insight to unique plant vulnerabilities not fully recognized in the safety analysis report Design Basis Accident or Transient analyses. It is the intent of this criterion that those requirements that 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 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.

3. PRA INSIGHTS

Introduction and Objectives

Reference 3 includes a statement that 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 these 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 a variety of PRA material with two purposes: 1) to identify if a Specification component or topic is addressed by PRA, and 2) if addressed, to judge if the Relocated Specification component or topic is risk-important. The intent of the PRA review was to provide an additional screen to the deterministic criteria. This review was accomplished in the generic Babcock and Wilcox Owners Group submittal BAW-1923, Volume I and B&W Owners Group Technical Report 47-1170689-00 (Reference 1). The results of this generic review have been confirmed by Duke Energy for the applicable ONS Units 1, 2 and 3 Specifications to be relocated. Where Reference 1 did not review a ONS Units 1, 2 and 3 Technical Specification against the criteria of Reference 3, Duke Energy performed a review similar (but not identical) to that described below for Reference 1.

Assumptions and Approach

Any relocated system or component specifically addressed by PRA 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. PRA data, initiating events, sequence frequencies, fault trees, and event trees were examined to aid in the judgement of the risk significance. No specific screening criteria were relied upon to make the decision for risk significance. In some case the judgements were clearly supported by the PRA material used. In other cases the judgements were subjective. The assessment was based on available literature on plant risk insights and PRAs. Table 3-1 lists the PRAs used for making the assessments and is provided at the end of this section.

When making judgments based on PRA, the general approach used was to assume a loss or degradation of the function for those systems or components of the relocated specifications. In one sense this provides a crude sensitivity analysis to permit judgements on the importance of the subject of the specification under review. This approach is conservative since the related specifications will be managed by Duke Energy to prevent significant degradation of system performance.

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

Table 3-1
PRA Material Used

1. NUREG 1050, "Probabilistic Risk Assessment (PRA) Reference Document," September 1984, U.S. Nuclear Regulatory Commission

2. NUREG/CR-3762, EGG - 2311, "Identification of Equipment and Components Predicted as Significant Contributors to Severe Core Damage," May 1984, EG&G Idaho, Inc.

3. NSAC/60-SY, "Oconee PRA, A Probabilistic Risk Assessment of Oconee Unit 3," 6/84, co-sponsored by Duke Power Company and the Nuclear Safety Analysis Center of the Electric Power Research Institute.

4. "Midland Nuclear Plant Probabilistic Risk Assessment," Consumers Power Company, entered into public docket by letter dated May 7, 1984, Docket Nos 50-329, 50-330

5. NUREG/CR-2787, SAND 82-0978, "Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One - Unit 1 Nuclear Power Plant." 6/82, Prepared by Sandia Labs for the U.S. Nuclear Regulatory Commission.

6. NSAC 84. "Zion Nuclear Plant Residual Heat Removal PRA," July 1985, Nuclear Safety Analysis Center, Electric Power Research Institute.

4. RESULTS OF APPLICATION OF SCREENING CRITERIA

The screening criteria from Section 2 were applied to the ONS Units 1, 2 and 3 Technical Specifications. The following Summary Disposition Matrix is a summary of that application indicating which Specifications are being retained or relocated, the criteria for inclusion, if applicable, the NRC results of the criteria application as expressed in the NRC Staff Review of NSSS Vendor Owners Groups Application of The Commission's Interim Policy Statement Criteria To Standard Technical Specifications, Wilgus/Murley letter dated May 9, 1988, and any necessary explanatory notes. Discussions that document the rationale for the relocation of each Specification which failed to meet the screening criteria are provided in Appendix A, except as noted in the Summary Disposition Matrix.

5. REFERENCES

1. 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

2. NUREG-1430, "Standard Technical Specifications, Babcock & Wilcox Plants," Revision 1, April 1995.

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

CURRENT TS	CURRENT DESCRIPTION	NEW TS NUMBER	RETAINED CRITERION FOR INCLUSION	NOTES
1	DEFINITIONS	1.1		This section provides definitions for several defined terms used throughout the remainder of Technical Specifications. They are provided to improve the meaning of certain terms. As such, direct application of the Technical Specification 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 Technical Specifications.
2.1	SAFETY LIMITS, REACTOR CORE	2.1.1	Yes	Application of Technical Specification screening criteria is not appropriate. However, Safety Limits will be included in Technical Specifications as required by 10 CFR 50.36.
2.2	SAFETY LIMITS - REACTOR COOLANT SYSTEM PRESSURE	2.1.2	Yes	Same as above.
2.3	LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION	3.3.1	Yes-3	The 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 consequences of Design Basis Accidents and transients or are retained as directed by the NRC as the Functions are part of the RPS.
3.0	LIMITING CONDITION FOR OPERATION	3.0.3		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 of 3.0.3 will be retained in Technical Specifications, as modified consistent with NUREG-1430, Revision 1.
3.1	REACTOR COOLANT SYSTEM			
3.1.1	<u>Operational Components</u>			
3.1.1.a	Reactor Coolant Pumps	3.4.4	Yes-2	
3.1.1.b	Steam Generator	3.4.4	Yes-2	
3.1.1.c	Pressurizer Safety Valves	3.4.10	Yes-3	
3.1.2	<u>Pressurization, Heatup, and Cooldown Limitation</u>			
3.1.2.1	RCS Pressure & Heatup and Cooldown Limits	3.4.3	Yes-2	
3.1.2.2	ASME Leak Tests, Limits	3.4.3	Yes-2	

CURRENT TS	CURRENT DESCRIPTION	NEW TS NUMBER	RETAINED CRITERION FOR INCLUSION	NOTES
3.1.2.3	Connected Systems Leak Tests, Limits	3.4.3	Yes-2	
3.1.2.4	ASME Hydro Tests, Limits	3.4.3	Yes-2	
3.1.2.5	OTSG Secondary Pressure/Temperature Limits	Relocated		See Appendix A, page A-1.
3.1.2.6	Pressurizer Heatup/Cooldown, Spray Valve delta T Limits	Relocated		See Appendix A, page A-2.
3.1.2.7, 3.1.2.8	Not Used Not Used			
3.1.2.9	LTOP	3.4.12	Yes-2	
3.1.3	<u>Minimum Conditions for Criticality</u>			
3.1.3.1	RCS Temperature	3.4.2	Yes-2	
3.1.3.2	RCS Temperature	3.4.2	Yes-2	
3.1.3.3	RCS Reactivity Limit - Temperature	Deleted		Deleted, see Discussion of Changes for Section 3.4.
3.1.3.4	RCS Reactivity Limit - Pressurizer Conditions	3.4.9	Yes-2	
3.1.3.5	Safety Rod Position Limits	3.1.5	Yes-2	
3.1.4	<u>RCS Activity</u>	3.4.11	Yes-2	
3.1.5	Not Used.			
3.1.6.1 - 3.1.6.8	<u>RCS Leakage</u>	3.4.13	Yes-1, 2	
3.1.6.9	RCS Returnable leakage Limits	Relocated		See Appendix A, Page A-4.
3.1.7	Moderator Temperature Coefficient of Reactivity	3.1.3	Yes-2	
3.1.8	Not Used			
3.1.9	Low Power Physics Testing Restrictions	3.1.8	Yes-1, 2, 3	
3.1.10	Not used.			
3.1.11	Shutdown Margin	3.1.1	Yes-2	

CURRENT TS	CURRENT DESCRIPTION	NEW TS NUMBER	RETAINED CRITERION FOR INCLUSION	NOTES
3.2	HIGH PRESSURE INJECTION AND CHEMICAL ADDITION SYSTEMS			
3.2.1	High Pressure Injection (HPI) System	Relocated		See Appendix A, page A-6.
3.2.2/Table 4.1-3 Item 6	Boric Acid Source In Addition to the Borated Water Storage Tank	Relocated		See Appendix A, page A-6.
3.3	EMERGENCY CORE COOLING, REACTOR BUILDING COOLING, REACTOR BUILDING SPRAY, AND LOW PRESSURE SERVICE WATER SYSTEMS			
3.3.1	High Pressure Injection (HPI) System	3.5.2	Yes-3	
3.3.2	Low Pressure Injection (LPI) System	3.5.3	Yes-3	
3.3.3	Core Flood Tank (CFT) System	3.5.1	Yes-3	
3.3.4	Borated Water Storage Tank (BWST)	3.5.4	Yes-3	
3.3.5	Reactor Building Cooling (RBC) System	3.6.5	Yes-3	
3.3.6	Reactor Building Spray (RBS) System	3.6.5	Yes-3	
3.3.7	Low Pressure Service Water (LPSW) System	3.7.7	Yes-3	
3.4	SECONDARY SYSTEM DECAY HEAT REMOVAL			
3.4.1	Emergency Feedwater (EFW) Pumps and Flow Paths	3.7.5	Yes-3	
3.4.2	EFW Initiation Circuitry	3.3.14	Yes-3	
3.4.3	EFW Pumps and Flow Paths (Conditions/ Required Actions and Completion Times	3.7.5	Yes-3	
3.4.4	Main Steam Safety Relief Valves	3.7.1	Yes-3	
3.4.5	Secondary System Water Inventory	3.7.6	Yes-3	
3.4.6	Independence of EFW Controls and Integrated Control System	Relocated		See Appendix A, page A-8.

CURRENT TS	CURRENT DESCRIPTION	NEW TS NUMBER	RETAINED CRITERION FOR INCLUSION	NOTES
3.5	INSTRUMENTATION SYSTEMS			
3.5.1	Operation Safety Instrumentation RPS,	3.3.1 through 3.3.3	Yes-3	
	ESF,	3.3.5 through 3.3.8	Yes-3	
	CRD Breakers & SCR Control Relays	3.3.4	Yes-3	
3.5.2	Control Rod Group and Power Distribution Limits			
3.5.2.1	Shutdown Margin	3.1.1	Yes-2	
3.5.2.2.a 3.5.2.2.b.1 - 3.5.2.2.b.5	Movable Control Assemblies (CRA)	3.1.4	Yes-2	
3.5.2.2.b.6/ 4.7.2	Rod Program Verification	Relocated		See Appendix A, Page A-9.
3.5.2.2.c, d & e	Movable Control Assemblies (CRA)	3.1.4	Yes-2	
3.5.2.3	Worths of Single Inserted CRA	Deleted		Deleted, see Discussion of Changes for Section 3.1.
3.5.2.4	Quadrant Power Tilt	3.2.3	Yes-2	
3.5.2.5	Control Rod Positions	3.2.1	Yes-2	
3.5.2.6	Reactor Power Imbalance	3.2.2	Yes-2	
3.5.2.7	CRD Patch Panel	Relocated		See Appendix A, page A-9.
3.5.3	ESF Actuation Setpoints	3.3.5	Yes-3	
3.5.4/Table 4.1-1 Item 34	Incore Instrumentation	Relocated		See Appendix A, page A-11.
3.5.6	Accident Monitoring Instrumentation	3.3.8	Yes-3	
3.6	REACTOR BUILDING			
3.6.1 3.6.2 3.6.3	Containment Integrity	3.6.1	Yes-3	

CURRENT TS	CURRENT DESCRIPTION	NEW TS NUMBER	RETAINED CRITERION FOR INCLUSION	NOTES
3.6.4	Reactor Building Internal Pressure	3.6.4	Yes-3	
3.6.5	Manual Isolation Valves	3.6.3	Yes-3	
3.6.6	Leakage Limits	3.6.1	Yes-3	
3.7.0	None	3.0.4		This Specification provides generic guidance applicable to one or more Specifications in CTS Section 3.7. 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 of 3.0.4 will be retained and made applicable to the remainder of the Technical Specifications, as modified consistent with NUREG-1430, Revision 1.
3.7.1	AC Sources - Operating	3.8.1	Yes-3	
3.7.2	Distribution Systems - Operating	3.8.8	Yes-3	
3.7.3	EPSL Automatic Transfer Functions	3.3.17	Yes-3	
3.7.4	EPSL Voltage Sensing Circuits	3.3.18	Yes-3	
3.7.5	EPSL Keowee Emergency Start Function	3.3.21	Yes-3	
3.7.6	EPSL Degraded Grid Voltage Protection	3.3.19	Yes-3	
3.7.7	EPSL CT-5 Degraded Grid Voltage Protection	3.3.20	Yes-3	
3.7.8	DC Sources - Operating	3.8.3	Yes-3	
3.7.9	Vital Inverters - Operating	3.8.6	Yes-3	
3.7.10	Battery Cell Parameters	3.8.5	Yes-3	
3.8	FUEL MOVEMENT AND STORAGE IN THE SPENT FUEL POOL			
3.8.1	Radiation Monitoring in RB refueling Area	Relocated		See Appendix A, page A-13.
3.8.2	Core Flux Monitoring	3.9.2	Yes-3	

CURRENT TS	CURRENT DESCRIPTION	NEW TS NUMBER	RETAINED CRITERION FOR INCLUSION	NOTES
3.8.3	LPI pump and Cooler	3.9.4 & 3.9.5	Yes-4	Although the DHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk reduction. Therefore, the DHR System is retained as a Specification.
3.8.4	Boron Concentration	3.9.1	Yes-2	
3.8.5	Communication	Relocated		See Appendix A, page A-15.
3.8.6	Personnel Airlocks and Equipment Hatch	3.9.3	Yes-3	
3.8.7	Isolation Valves	3.9.3	Yes-3	
3.8.8	Fuel Assembly Separation; Auxiliary Hoist	Relocated		See Appendix A, page A-17.
3.8.9	Fuel Loading and Refueling Conditions Not Met	3.9.1 through 3.9.5	Yes-2, 3	
3.8.10	RB Purge and Radiation Monitor	3.9.3	Yes-3	
3.8.11	Minimum After Shutdown	Deleted		Deleted, see Discussion of Changes for Section 3.9.
3.8.12/4.14	SFP Ventilation	Relocated		See Appendix A, page A-19.
3.8.13	Minimum decay time for SF shipping and dry storage cask movement in SFP	3.7.15	Yes-2	
3.8.14	Suspended load movement restrictions over spent Fuel in SFP	Relocated		See Appendix A, page A-21.
3.8.15	SFP Boron Concentration	3.7.12	Yes-2	
3.8.16	SFP storage restrictions	3.7.13	Yes-2	
3.8.17	SFP Boron Concentration or Storage Locations Requirements Not Met	3.7.12 & 3.7.13	Yes-2	
3.9	LIQUID HOLDUP TANKS	5.5.13	Yes	Although this Specification does not meet any Technical Specification screening criteria, it has been retained in accordance with the NRC letter from W. T. Russell to the industry ITS Chairpersons, dated October 25, 1993.
3.10	GAS STORAGE TANK AND EXPLOSIVE GAS MIXTURE	5.5.13	Yes	Same as above.

CURRENT TS	CURRENT DESCRIPTION	NEW TS NUMBER	RETAINED CRITERION FOR INCLUSION	NOTES
3.12	RB POLAR CRANE AND AUXILIARY HOIST	Relocated		See Appendix A, page A-23.
3.13	SECONDARY SYSTEM ACTIVITY	3.7.14	Yes-2	
3.14/4.18	SNUBBERS	Deleted		Deleted, see Discussion of Changes for Section 3.7.
3.15	CONTROL ROOM PRESSURIZATION AND FILTERING SYSTEM AND PENETRATION ROOM VENTILATION SYSTEM	3.7.9 and 3.7.10	Yes-3	
3.16	CONTAINMENT HYDROGEN CONTROL SYSTEMS	Deleted		Deleted, see Discussion of Changes for Section 3.6.
3.17	Not Used			
3.18	STANDBY SHUTDOWN FACILITY	3.10	Yes-4	
4.0.1	Operational Modes	SR 3.0.1	Yes	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 of 3.0/4.0 will be retained in Technical Specifications, as modified consistent with NUREG-1430, Revision 1.
4.0.2	Time of Performance	SR 3.0.2	Yes	Same as above.
4.0.3	Entry into Operational Modes	SR 3.0.4	Yes	Same as above.
4.0.4	ASME Code Class 1,2, 3 Components	5.5.6	Yes	This Specification is actually a Surveillance Requirement which has been retained in the Administrative Controls programs for IST.
Table 4.1-1 Items 22, 25a, 25b, 27, 31, 32, 33, 35, 36, 38, 40 & 50	Instrument Surveillance Requirements	Relocated		See Appendix A, Page A-25. Split Criteria applied to implied LCOs. There are no explicit CTS LCO requirements associated with these surveillance requirements.
Table 4.1-2 Item 8	HPSW Pumps and Power Supplies	Relocated		See Appendix A, page A-27
Table 4.1-2 Item 9	Spent Fuel Cooling System	Relocated		See Appendix A, Page A-29. Split Criteria applied to implied LCO. There is no explicit CTS LCO requirement associated with this surveillance requirement.

CURRENT TS	CURRENT DESCRIPTION	NEW TS NUMBER	RETAINED CRITERION FOR INCLUSION	NOTES
4.16	Radioactive Material Sources	Relocated		See Appendix A, Page A-31. Split Criteria applied to implied LCO. There is no explicit CTS LCO requirement associated with this surveillance requirement.
5	DESIGN FEATURES	4.0		
5.1	SITE	4.1	Yes	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Design Features will be included in Technical Specifications as required by 10 CFR 50.36.
5.2	CONTAINMENT	N/A	No	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Design Features will be included in Technical Specifications as required by 10 CFR 50.36.

CURRENT TS	CURRENT DESCRIPTION	NEW TS NUMBER	RETAINED CRITERION FOR INCLUSION	NOTES
6	ADMINISTRATIVE CONTROLS	5.0		
6.1	ORGANIZATION, REVIEW, AND AUDIT	5.1, 5.2 and 5.3	Yes	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.
6.2	ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE OCCURRENCE	N/A	No	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.
6.3	ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED	2.2	Yes	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.
6.4	STATION OPERATING PROCEDURES	5.4	Yes	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.
6.5	STATION OPERATING RECORDS	N/A	No	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.
6.6	STATION REPORTING REQUIREMENTS	5.6	Yes	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.
6.7	ENVIRONMENTAL QUALIFICATION	N/A	No	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.
6.8	Not Used			
6.9	CORE OPERATING LIMITS REPORT	5.6.5	Yes	Application of Technical Specification screening criteria is not appropriate. However, specific portions of Administrative Controls will be included in Technical Specifications as required by 10 CFR 50.36.

3.1.2.5 STEAM GENERATOR P/T LIMITS

LCO Statement:

3.1.2.5 The secondary side of the steam generator shall not be pressurized above 237 psig if the temperature of the vessel shell is below 110°F.

Discussion:

The limitations on steam generator pressure and temperature provide protection against non-ductile failure of the secondary side (shell) of the steam generator. These limits are calculated using ASME code for Class A components and are considered to be conservative.

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 As discussed in Section 4.0 (Appendix A, page A-73) and summarized in the Disposition Matrix of B&W Owners Group Technical Report 47-1170689-0, Steam Generator P/T Limits were found to be a non-significant risk contributor to core damage frequency and plant risk. Duke Energy has reviewed this evaluation, considers it applicable to ONS Units 1, 2 and 3, and concurs with the assessment.

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.

3.1.2.6 PRESSURIZER HEATUP AND COOLDOWN LIMITS and SPRAY VALVE ΔT LIMITS

LCO Statement:

3.1.2.6 The pressurizer heatup and cooldown rates shall not exceed 100°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 410°F.

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.

Comparison to Screening Criteria:

Criterion 1 Pressurizer heatup and cooldown rates and 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 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 As discussed in Section 4.0 (Appendix A, page A-59) and summarized in the Disposition Matrix of B&W Owners Group Technical Report 47-1170689-0, the Pressurizer heatup and cooldown rates and temperature limitations were found to be a non-significant risk contributor to core damage frequency and plant risk. Duke Energy has reviewed this evaluation, considers it applicable to ONS Units 1, 2 and 3, and concurs with the assessment.

Conclusion:

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

3.1.6.9 RCS Returnable Leakage Limits

LCO Statement:

3.1.6.9 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1, 3.1.6.2, 3.1.6.3, 3.1.6.4, 3.1.6.5, 3.1.6.6 or 3.1.6.7 except that such losses when added to leakage shall not exceed 30 gpm.

Discussion:

The upper limit of 30 gpm is based on the contingency of a complete loss of station power. A 30 gpm loss of water in conjunction with a complete loss of station power and subsequent cooldown of the reactor coolant system by the turbine bypass system (set at 1,040 psia) and steam driven emergency feedwater pump would require more than 60 minutes to empty the pressurizer from the combined effect of system leakage and contraction. This will be ample time to restore electrical power to the station and makeup flow to the reactor coolant system.

Comparison to Screening Criteria:

Criterion 1 RCS returnable leakage limits 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 RCS returnable leakage 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.

Criterion 3 RCS returnable leakage 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 RCS returnable leakage limits are not addressed in B&W Owners Group Technical Report 47-1170689-0. No ONS PSA risk measure or insight indicate the returnable RCS leakage limits are significant to public health or safety.

Conclusion:

Since the screening criteria have not been satisfied, the RCS returnable leakage limit requirements may be relocated to licensee controlled documents outside the Technical Specifications.

3.2 HIGH PRESSURE INJECTION AND CHEMICAL ADDITION SYSTEMS

LCO Statement:

The reactor shall not be critical unless the following conditions are met:

3.2.1 Two high pressure injection pumps per unit are operable except as specified in 3.3.

3.2.2 One source per unit of concentrated soluble boric acid in addition to the borated water storage tank is available and operable.

This source will be the concentrated boric acid storage tank with the volume and boron concentration within the limits of the Core Operating Limits Report with a temperature at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the high pressure injection system shall be operable and shall have the same temperature requirement as the concentrated boric acid storage tank. At least one channel of heat tracing capable of meeting the above temperature requirement shall be in operation. One associated boric acid pump shall be operable.

If the concentrated boric acid storage tank with its associated flowpath is unavailable, but the borated water storage tank is available and operable, the concentrated boric acid storage tank shall be restored to operability within 72 hours or the reactor shall be placed in a hot shutdown condition and be borated to a shutdown margin equivalent to 1% $\Delta k/k$ at 200°F within the next twelve hours; if the concentrated boric acid storage tank has not been restored to operability within the next 7 days the reactor shall be placed in a cold shutdown condition within an additional 30 hours.

If the concentrated boric acid storage tank is available but the borated water storage tank is neither available nor operable, the borated water storage tank shall be restored to operability within one hour or the reactor shall be placed in a hot shutdown condition within 6 hours and in a cold shutdown condition within an additional 30 hours.

Discussion:

The High Pressure Injection and Chemical Addition Systems ensure negative reactivity control is available for normal operation (normal makeup and chemical shim reactivity control). HPI with boron addition from the CBAST is an alternative method for emergency boration of the RCS in the event of stuck control rods following a reactor trip. The primary method for emergency boration is HPI using borated water from the BWST. The reactivity control capability provided by the combination of HPI and CBAST is not assumed to mitigate any design basis accident or transient as sources of borated water are assumed in the safety analysis. The requirements for the High Pressure Injection System capability with regard to the Borated Water Storage tank is included in a specification 3.3.

Comparison to Screening Criteria:

Criterion 1 The chemical addition 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 chemical addition 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 chemical addition 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 As discussed in Section 4.0 (Appendix A, pages A-3 and A-7) and summarized in the Disposition Matrix of B&W Owners Group Technical Report 47-1170689-0, the chemical addition system was found to be a non-significant risk contributor to core damage frequency and plant risk. Duke Energy has reviewed this evaluation, considers it applicable to ONS Units 1, 2 and 3, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the sources of boric acid solution requirements may be relocated to licensee controlled documents outside the Technical Specifications.

Appendix A

3.4.6 INDEPENDENCE OF EFW CONTROLS AND INTEGRATED CONTROL SYSTEM (ICS)

LCO Statement:

3.4.6 The controls of the emergency feedwater system shall be independent of the Integrated Control System.

Discussion:

The independence of EFW controls from ICS is a requirement placed upon plant design and is not within the control of the plant operators.

Comparison to Screening Criteria:

Criterion 1 The independence of EFW Controls from ICS 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 independence of EFW Controls from ICS 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 independence of EFW Controls from ICS 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 Independence of EFW Controls from ICS is not addressed in B&W Owners Group Technical Report 47-1170689-0. No ONS PSA risk measure or insight indicates the independence of EFW Controls from ICS is significant to public health or safety.

Conclusion:

Since the screening criteria have not been satisfied, the independence of EFW Controls requirement may be relocated to licensee controlled documents outside the Technical Specifications.

3.5.2.7 CONTROL ROD DRIVE PATCH PANEL

LCO Statements:

3.5.2.2.b A control rod shall be declared inoperable if the following condition exist for that rod: . . .

6. The control rod does not meet the rod program verification of Specification 4.7.2.

3.5.2.7 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the manager or his designated alternate.

Discussion:

The control rod drive patch panels are a feature of the CRDM power supplies to provide flexibility in establishing the value of rod worth between rod groups. It is possible to patch (i.e., program) any rod into any group with the exception of Group 8. These panels are in two cabinets which are kept locked due to the sensitivity of their function. The control rod program ensures the control rods are programmed to operate in the core position and rod group consistent with the core licensing analysis. The locked or unlocked status of the CRD patch panels is not assumed in the safety analyses, nor does the panel lock mechanisms serve an accident mitigation function.

Comparison to Screening Criteria:

Criterion 1 Control rod programming 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 Control rod programming 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 Control rod programming 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 As discussed in Section 4.0 (Appendix A, page A-13) and summarized in the Disposition Matrix of B&W Owners Group Technical Report 47-1170689-0, control rod programming was found to be a non-significant risk contributor to core damage frequency and plant risk. Duke Energy has reviewed this evaluation, considers it applicable to ONS Units 1, 2 and 3, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the control rod programming requirements may be relocated to a licensee controlled document outside the Technical Specifications.

3.5.4 INCORE INSTRUMENTATION

LCO Statement:

3.5.4.1 At or above 80 percent of the power allowable for the existing reactor coolant pump operating combination, incore detectors shall be operable as necessary to meet the following:

a. For axial imbalance measurements:

At least three detectors in each of at least three strings shall lie in the same axial plane, with one plane in each axial core half. The axial planes in each core half shall be symmetrical about the core mid-plane. The detector strings shall not have radial symmetry.

b. For quadrant power tilt measurements:

At least two sets of at least four detectors shall lie in each axial core half. Each set of detectors shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane. Detectors in the same plane shall have quarter core radial symmetry.

3.5.4.2 If requirements of 3.5.4.1 are not met, power shall be reduced below 80 percent of the power allowable for the existing reactor coolant pump combination within eight hours and incore detector measurements shall not be used to determine axial imbalance or quadrant power tilt.

Discussion:

The incore detector system is used to provide detailed information on the reactor core neutron flux distribution. This information is used to verify that the axial power distribution and quadrant tilt are within their limits. The axial power distribution and quadrant tilt limits are established to help ensure that the maximum core power peaking assumed in the plant DBA is not exceeded. No automatic actions result from the incore detector system. The power range neutron flux instrumentation is also used to measure axial power distribution and quadrant power tilt. These detectors, however, provide a coarser measurement due to their location and the fewer number of detectors than that supplied by the incore detection system. The reactor protection system uses the power range neutron flux instruments to generate reactor trips due to unacceptable axial core power distribution by way of the flux/imbalance/RCS flow trip signal.

Comparison to Screening Criteria:

Criterion 1 The incore detector 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 incore detector 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 incore detector 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 As discussed in Section 4.0 (Appendix A, page A-29) and summarized in the Disposition Matrix of B&W Owners Group Technical Report 47-1170689-0, Incore Neutron Detectors were found to be a non-significant risk contributor to core damage frequency and plant risk. Duke Energy has reviewed this evaluation, considers it applicable to ONS Units 1, 2 and 3, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the incore detector system requirements may be relocated to other licensee controlled documents outside the Technical Specifications.

Appendix A

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 RIA-3 and by a portable bridge monitor for each bridge which is being used for fuel handling. Radiation levels in the spent fuel storage area shall be monitored by RIA-6 and by a portable bridge monitor. If any of these instruments becomes 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.

Discussion:

Radiation monitors RIA-3 and RIA-6 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. These monitors serve to notify personnel of an increase in radiation in these areas. When either is inoperable, the local radiation coverage and alarm functions are provided by portable survey instrumentation. Operability of these monitors is not an assumption in the safety analysis, nor do they serve any accident mitigation function.

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 are not addressed in B&W Owners Group Technical Report 47-1170689-0. No ONS PSA risk measure or insight indicates the radiation monitoring in the reactor building refueling area and spent fuel storage area is significant to public health or safety.

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.

Appendix A

3.8.5 COMMUNICATIONS

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.

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. This communication is not an assumption in the accident analyses, nor does it serve any accident mitigation function.

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 As discussed in Section 4.0 (Appendix A, page A-87) and summarized in the Disposition Matrix of B&W Owners Group Technical Report 47-1170689-0, Communications was found to be a non-significant risk contributor to core damage frequency and plant risk. Duke Energy has reviewed this evaluation, considers it applicable to ONS Units 1, 2 and 3, and concurs with the assessment.

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 FUEL ASSEMBLY SEPARATION AND USE OF AUXILIARY HOIST

LCO Statement:

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

Irradiated fuel assemblies may be handled with the Auxiliary Hoist provided no other irradiated fuel assembly is being handled in the fuel transfer canal.

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 the bridges or use of the auxiliary hoist.

Comparison to Screening Criteria:

Criterion 1 The separation requirement when moving irradiated fuel assemblies or use of the auxiliary hoist 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 The separation requirement when moving irradiated fuel assemblies or use of the auxiliary hoist 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 The separation requirement when moving irradiated fuel assemblies or use of the auxiliary hoist 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 As discussed in Section 4.0 (Appendix A, page A-89) and summarized in the Disposition Matrix of B&W Owners Group Technical Report 47-1170689-0, the Fuel Handling Bridge was found to be a non-significant risk contributor to core damage frequency and plant risk. Duke Energy has reviewed this evaluation, considers it applicable to ONS Units 1, 2 and 3, and concurs with the assessment.

Conclusion:

Since the screening criteria have not been satisfied, the separation requirement when moving irradiated fuel assemblies or use of the auxiliary hoist are may be relocated to other licensee controlled documents outside the Technical Specifications.

3.8.12 SPENT FUEL POOL VENTILATION

LCO Statement:

3.8.12 Two trains of spent fuel pool ventilation shall be operable with the following exceptions:

a. With one train of spent fuel pool ventilation inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the operable spent fuel pool ventilation train is in operation and discharging through the Reactor Building purge filters.

b. With no spent fuel pool ventilation filter operable, suspend all operations involving movement of fuel within the storage pool or crane operations with loads over the storage pool until at least one train of spent fuel pool ventilation is restored to operable status.

c. This specification does not apply during reracking operations with no fuel in the spent fuel pool.

Discussion:

The Spent Fuel Pool Ventilation System maintains a suitable environment in the Spent Fuel Pool area for the proper operation, maintenance and testing of equipment as well as for personnel access. In the filtered mode of operation, exhaust air is directed through the Reactor Building Purge Filter Train before being discharged to the unit vent, however, no credit is taken in the safety analyses for the filtration provided by these filters. The system is not required for nuclear safety and is not operational in the event of a loss of power. Offsite doses are within the guideline values of 10 CFR 100 without the benefit of operation of the Spent Fuel Pool Ventilation System.

Comparison to Screening Criteria:

Criterion 1 The Spent Fuel Pool Ventilation 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 Spent Fuel Pool Ventilation 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 Spent Fuel Pool Ventilation 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 Spent Fuel Pool Ventilation System is not addressed in B&W Owners Group Technical Report 47-1170689-0. No ONS PSA risk measure or insight indicates the Spent Fuel Pool Ventilation System is significant to public health or safety.

Conclusion:

Since the screening criteria have not been satisfied, The Spent Fuel Pool Ventilation System requirements may be relocated to other licensee controlled documents outside the Technical Specifications.

3.8.14 Suspended Loads Over Spent Fuel

LCO Statement:

3.8.14 No suspended loads of more than 3000 lbm shall be transported over spent fuel stored in either spent fuel pool.

Discussion:

This specification prohibits transporting loads greater than 3000 lbm a fuel assembly with a control rod and the associated fuel handling tool(s) over spent fuel stored in the spent fuel pool. This limitation was established to preclude movement of loads weighing more than a fuel assembly and a control rod over irradiated fuel stored in the spent fuel storage racks during construction activity associated with replacing fuel storage racks in 1979. This construction activity has been completed and the temporary crane which was used to transport the new racks in the SFP has been removed from the SFP area. The 100 ton crane used for routine cask handling operates over the spent fuel pool at only one end, and has access only to the cask loading pit and cask loading platform. The crane bridge and trolley hard stops prevent travel over any area where spent fuel is stored in the fuel racks.

Comparison to Screening Criteria:

Criterion 1 The restriction upon transporting loads over spent fuel in the spent fuel pool 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 severity for fuel handling accidents is limited by the limits placed upon transportation of loads over spent fuel. These are physical stops for the 100 ton crane. The temporary construction crane has been removed from the spent fuel pool. These limits are not process variables monitored or controlled by the operator. Therefore Criterion 2 is not satisfied.

Criterion 3 The restriction upon transporting loads over spent fuel in the spent fuel pool 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 upon transporting loads (Crane Travel) over spent fuel in the spent fuel pool is not a structure, system or component addressed in B&W Owners Group Technical Report 47-1170689-0. No ONS PSA risk measure or insight indicates the restriction upon transporting loads over spent fuel in the spent fuel pool is a significant risk contributor to core damage fuel pool is a significant risk contributor to core damage frequency or plant risk.

Conclusion:

Since the screening criteria have not been satisfied, the restriction upon transporting loads over spent fuel in the spent fuel pool may be relocated to other licensee controlled documents outside the Technical Specifications.

3.12 RB POLAR CRANE AND AUXILIARY HOIST

LCO Statement:

3.12.1 The reactor building polar crane shall not be operated over the fuel transfer canal when any fuel assembly is being moved.

3.12.2 The auxiliary hoist shall not be operated over the fuel transfer canal when any fuel assembly is being moved unless the hoist is being used to move that assembly.

3.12.3 During this period when the reactor vessel head is removed and irradiated fuel is in the reactor building and fuel is not being moved, the reactor building polar crane and auxiliary hoist shall be operated over the fuel transfer canal only where necessary and in accordance with approved operating procedures stating the purpose of such use.

3.12.4 When the reactor vessel head is removed and the polar crane is being operated in areas away from the fuel transfer canal, the flagman shall be located on top of the secondary shield wall when the polar crane hook is above the elevation of the fuel transfer canal.

3.12.5 During the period when the reactor coolant system is pressurized above 300 psig, and is above 200°F, and fuel is in the core, the reactor building polar crane shall not be operated over the steam generator compartments.

Discussion:

Applies to the use of the reactor building polar crane over the steam generator compartments and the fuel transfer canal and the auxiliary hoist over the fuel transfer canal. These restrictions preclude the dropping of materials or equipment into the reactor vessel and possibly damaging the fuel to the extent that an escape of fission products would result.

The fuel transfer canal is delineated by readily visible markers at an elevation above which the reactor building polar crane does not normally handle loads. Restriction in the use of the reactor building polar crane over the steam generator compartments is administratively controlled to preclude damage to the steam generators and the RCS system.

Comparison to Screening Criteria:

Criterion 1 The RB polar crane and auxiliary hoist 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 The RB polar crane and auxiliary hoist 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 The RB polar crane and auxiliary hoist 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 RB polar crane and auxiliary hoist are not addressed in B&W Owners Group Technical Report 47-1170689-0. No ONS PSA risk measure or insight indicates the RB polar crane and auxiliary hoist are significant to public health or safety.

Conclusion:

Since the screening criteria have not been satisfied, the RB polar crane and auxiliary hoist restrictions may be relocated to other licensee controlled documents outside the Technical Specifications.

Appendix A

TABLE 4.1-1 INSTRUMENT SURVEILLANCE REQUIREMENTS

LCO Statement:

The following instrument channel surveillance requirements from Table 4.1-1 imply that associated LCOs exist. However, unique LCOs associated with these surveillance requirements are not specifically identified in Section 3 of the CTS:

Item #	Channel Description
22.	Pressurizer Temperature
25a.	Core Flood Tank Pressure
25b.	Core Flood Tank Level
27.	Letdown Storage Tank Level
31a.	Boric Acid Mix Tank Level
31b.	Boric Acid Mix Tank Temperature
32a.	Concentrated Boric Acid Storage Tank Level
32b.	Concentrated Boric Acid Storage Tank Temperature
33.	Containment Temperature
35.	Emergency Plant Radiation Instruments
36.	Environmental Monitors
38.	Reactor Building Emergency Sump Level
40.	Turbine Overspeed Trip
50.	PORV and Safety Valve Position Indicators

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 for which there is no corresponding LCO in CTS section 3. A comparison of the table with the LCOs resulted in 14 items being identified as not having a match.

Comparison to Screening Criteria:

Criterion 1 The listed instruments 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 The listed instruments 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 The listed instruments 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 listed instruments are not addressed in the ONS 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, requirements associated with the listed instruments may be relocated to other licensee controlled documents outside the Technical Specifications.

TABLE 4.1-2 High Pressure Service Water Pumps and Power Supplies

LCO Statement:

The following Surveillance Requirement from Table 4.1-2 implies an LCO exists. CTS Table 4.1-2 list a surveillance requirement for which there is no corresponding LCO in CTS section 3.

Item #	Description
8.	High Pressure Service Water Pumps and Power Supplies Monthly Functional Test

Discussion:

This High Pressure Service Water pumps are used primarily for fire protection throughout the Oconee station. In the event of a loss of the normal LPSW supply, the HPSW system automatically supplies cooling water to the HPI pump motor coolers. For loss of AC power, HPSW via the elevated water storage tank automatically supplies cooling water to the turbine driven emergency feedwater pump and its associated oil cooler, and maintains CCW pump bearing cooling water and cooling water for the CCW pump motors.

Comparison to Screening Criteria:

Criterion 1 The High Pressure Service Water pumps 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 The High Pressure Service Water pumps 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 The High Pressure Service Water pumps 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 High Pressure Service Water System is addressed in the ONS PSA the HPSW pumps are not safety significant since at several hours of water is available from the Elevated Water Storage Tank without operation of the pumps. The HPSW pumps are not credited in any 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 High Pressure Service Water pump requirements may be relocated to other licensee controlled documents outside the Technical Specifications.

Appendix A

TABLE 4.1-2 SPENT FUEL POOL COOLING SYSTEM

LCO Statement:

The following Surveillance Requirement from Table 4.1-2 implies an LCO exists. CTS Table 4.1-2 list a surveillance requirement for which there is no corresponding LCO in CTS section 3.

Item #	Description
9.	Spent Fuel Pool Cooling System Functional

Discussion:

The spent fuel cooling system provides decay heat removal for the spent fuel stored in the spent fuel pool. Other system functions are to maintain spent fuel pool inventory, clarity and chemistry within acceptable levels.

Comparison to Screening Criteria:

Criterion 1 The Spent Fuel Pool Cooling 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 Spent Fuel Pool Cooling 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 Spent Fuel Pool Cooling 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 Spent Fuel Pool Cooling System is not addressed in the ONS PSA and are not credited in any 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 Spent Fuel Cooling System requirements may be relocated to other licensee controlled documents outside the Technical Specifications.

4.16 Radioactive Material Sources

LCO Statement:

CTS 4.16 imposes a Surveillance Requirement which implies an LCO exists. CTS 4.16, "Radioactive Material Sources" specifies surveillance requirements for which there is no corresponding LCO in CTS section 3.

Discussion:

This specification requires leakage testing for sealed sources containing radioactive material in non gaseous form, other than tritium with a half life greater than 30 days. This specification assures that leakage from byproduct, source and special nuclear material seal sources do not exceed allowable limits. Sealed sources are exempt when the source contains $\leq 100\ \mu Ci$ of beta and/or gamma emitting material or $\leq 10\ \mu Ci$ of alpha emitting material.

Comparison to Screening Criteria:

Criterion 1 Sealed Source Contamination 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 Sealed Source Contamination 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 Sealed Source Contamination 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 PSA does not address sealed sources.

Conclusion:

Since the screening criteria have not been satisfied, requirements associated with Sealed Source leakage requirements may be relocated to other licensee controlled documents outside the Technical Specifications.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 1.0 - USE AND APPLICATION

ATTACHMENT 1

TECHNICAL SPECIFICATIONS

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 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 control components with part length absorbers 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 the entire channel, including the required sensor, alarm, display, and trip functions, and shall include 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 Definitions

CHANNEL CALIBRATION (continued)	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.
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, including required alarms, interlocks, display, and trip functions. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the channel is functionally tested.
CONTROL RODS	CONTROL RODS shall be all full length safety and regulating rods.
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 unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
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

(continued)

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)

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."

$\bar{E}$ – 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 > 30 minutes, making up at least 95% of the total noniodine activity in the coolant.

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;

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. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE.

c. Pressure Boundary LEAKAGE

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

(continued)

1.1 Definitions (continued)

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.
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 UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.

(continued)

1.1 Definitions (continued)

QUADRANT POWER TILT (QPT)	QPT shall be defined by the following equation and is expressed as a percentage. $$QPT = 100\left(\frac{\text{Power in any Core Quadrant}}{\text{Average Power of all Quadrants}} - 1\right)$$
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2568 MWt.
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.

(continued)

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	≥ 250
4	Hot Shutdown[b]	< 0.99	NA	250 > T > 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 not 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.

(continued)

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)

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.	24 hours 24 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. 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)

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.	Once per 8 hours
	OR	
	A.2 Place the channel in bypass.	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

(continued)

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.

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.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 1.0 - USE AND APPLICATION

ATTACHMENT 2

BASES

There are no bases associated with ITS Section 1.0.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 1.0 - USE AND APPLICATION

ATTACHMENT 3

CTS MARKUP AND DISCUSSION OF CHANGES

A1 <except as marked>

Specification 1.1

~~TECHNICAL SPECIFICATIONS~~

~~1~~ ~~DEFINITIONS~~

NOTE
The defined terms of this section appear in capitalized typeand are applicable thoughout these Technical Specifications and Buses

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

RATED THERMAL POWER

~~1.1~~ RATED Thermal POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant. A3

~~Rated power is defined as a steady state reactor core output~~ of 2568 MWt.

~~1.2~~ ~~REACTOR OPERATING CONDITIONS~~

MODE 5

~~1.2.1~~ Cold Shutdown

The reactor is in the cold shutdown condition when it is subcritical by at least ~~1 percent Δk/k~~ 0.99 Keff and T_{avg} is no more than 200°F. ~~Pressure is defined by Specification 3.1.2.~~ A4 A25 A18

Average Reactor Coolant Temperature

MODE 3

~~1.2.2~~ Hot ~~Shutdown~~ standby

The reactor is in the hot shutdown condition when it is subcritical by at least ~~1 percent Δk/k~~ 0.99 Keff and T_{avg} is ~~at or greater than 525°F.~~ ≥ 250 A4 A6

~~1.2.3~~ ~~Reactor Critical~~

~~The reactor is critical when the neutron chain reaction is self-sustaining and K_{eff} = 1.0.~~ A4

MODE 2

~~1.2.4~~ ~~Hot Standby~~ Startup Apply table 1.1-1, Note (a) A7 A4

The reactor is in the hot standby condition when all of the following conditions exist: A18

Average Reactor coolant temperature

a. T_{avg} is greater than 525°F. A19

b. The ~~reactor is critical~~. reactivity condition ≥ 0.99 Keff

c. ~~Indicated neutron power on the power range channels~~ is less than ~~2~~ 5 percent of ~~rated power~~ RTP. L4 A13

MODE 1

~~1.2.5~~ Power Operation Apply table 1.1-1, Note (a) A4 A19

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~~. L4

MODE 6

~~1.2.6~~ Refueling ~~Shutdown~~ A13

The reactor is in the refueling shutdown condition when ~~even with all rods removed, the reactor would be subcritical by at least 1 percent Δk/k and the coolant temperature at the low pressure injection pump suction is no more than 140°F. Pressure is defined by Specification 3.1.2.~~ A refueling shutdown refers to a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods. A4 M1 A25 A14

One or more RV head bolts less than Fully 1-1 tensioned

Table 1.1-1 Note(c)

7/19/74 M1

Add MODE 4 A20

Add Table 1.1-1 Note 6 for MODES 4 + 5 M1

A1 <except as marked> M2

Movement of any fuel, sources or reactivity control components within the reactor vessel

and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position. A21

CORE ALTERATION

1.2.7 ~~Refueling Operation~~ CORE ALTERATION

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

~~1.2.8~~ Startup A13

MODE 2

The reactor shall be considered in the ~~startup~~ mode 2 when the ~~shutdown margin is reduced with the intent of going critical.~~ reactivity condition is ≥ 0.99 and the RATED THERMAL POWER is ≤ 5%. <Apply Table 1.1-1; Note a> A4 L3

1.3 OPERABLE - OPERABILITY

OPERABLE-OPERABILITY

A system, subsystem, train, component or device shall be ~~considered~~ OPERABLE <or have OPERABILITY> when it is capable of performing its specified ~~intended~~ safety functions. Implicit in this definition shall be the assumption that all essential auxiliary equipment required in order to assure performance of the safety function is capable of performing its related support function(s). Auxiliary equipment includes but is not limited to normal or emergency electrical power sources, cooling and seal water, lubrication, instrumentation and controls, etc. If either the normal or emergency power to system, subsystem, train, component or device is not available it is considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (a) the alternate power source is available, and (b) the redundant system is operable. <See 3.8> <See 3.0>

1.4 PROTECTIVE INSTRUMENTATION LOGIC

1.4.1 Instrument Channel

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 in nature.

LA2

1.4.2 Reactor Protective System

The reactor protective system is shown in Figures 7.2-1 and 7.2-4 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 protective channels, their associated instrument channel inputs, manual trip switch, all rod drive protective trip breakers and activating relays or coils.

1.4.3 Protective Channel

A protective channel as shown in Figure 7.2-1 of the FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply units, amplifiers and bistable modules provided for every reactor protective safety parameter) is a combination of instrument channels forming a single digital output to the protective system's coincidence logic. It includes a shutdown bypass circuit, a protective channel bypass circuit and reactor trip module and provision for insertion of a dummy bistable.

A1 <except as marked>

LA2

~~1.4.4 Reactor Protective System Logic~~

~~This system utilizes reactor trip module relays (coils and contacts) in all four of the protective channels as shown in Figure 7.2-1 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 protective channels.~~

~~1.4.5 Engineered 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 7.3-1 of the FSAR. The logic sub-system is wired to provide appropriate signals for the actuation of redundant Engineered Safety Features equipment on a two-of-three basis for any given parameter.~~

~~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.~~

A2

~~1.5 INSTRUMENTATION SURVEILLANCE~~

~~1.5.1 Trip Test~~

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

A9

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the channel is functionally tested

A27

CHANNEL FUNCTIONAL TEST

~~1.5.2~~ Channel FUNCTIONAL Test <CHANNEL FUNCTIONAL TEST definition as depicted in the ITS>

A channel test is the injection of an internal or external test signal into the channel (as close to the sensor as practicable) to verify its proper output response; including alarm and/or trip initiating action where applicable.

or actual

L1

CHANNEL CHECK

~~1.5.3~~ ~~Instrument~~ Channel Check <CHANNEL CHECK definition per ITS>

M3

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.

CHANNEL CALIBRATION

~~1.5.4~~ ~~Instrument~~ Channel Calibration <CHANNEL CALIBRATION as depicted in the ITS>

An instrument channel calibration is a test, and adjustment (if necessary), to establish that the channel output responds with ~~acceptable~~ necessary 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~~ the required sensor actuation, alarm, display or trip and shall be deemed to include the channel Functional test.

A23

L2

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.

A24

A 139/139/136
5/30/85

The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps so the entire channel is calibrated.

A1 <except as marked>

1.5.5 Heat Balance Check

A heat balance check is a comparison of the indicated neutron power and core thermal power. LA2

1.5.6 Heat Balance Calibration

An adjustment of the power range channel amplifiers output to agree with the core thermal power as determined by a heat balance on the secondary side of the steam generator considering all heat losses and additions. LA2

~~1.5.~~7 Staggered Test Basis

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.

~~1.6 POWER DISTRIBUTION~~

1.6.1 Quadrant Power Tilt (QPT)

QUADRANT POWER TILT (QPT)

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

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

1.6.2 AXIAL ~~Reactor~~ Power Imbalance

AXIAL POWER IMBALANCE

Axial ~~Reactor~~ power imbalance is 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 power. 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. A25

1.7 CONTAINMENT INTEGRITY

Containment integrity exists when the following conditions are satisfied: <See 3.6>

a. The equipment hatch is closed and sealed and both doors of the personnel hatch and emergency hatch are closed and sealed except as in b below.

b. At least one door of the personnel hatch and the emergency hatch is closed and sealed during refueling or during personnel passage through these hatches.

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

d. All automatic containment isolation valves are operable or locked closed.

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

4.7 ?

A1 <except as marked>

~~1.8 RADIOLOGICAL EFFLUENT CONTROL~~

1.8.1 Source Check

A Source Check is the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

A2

1.8.2 Offsite Dose Calculation Manual (ODCM)

The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Environmental Radiological Monitoring Program. FSAR Chapter 16 shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Sections 6.4.6 and 6.4.7 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.6.1.4 and 6.6.1.5.

<SEE 5.0>

1.8.3 Process Control Program (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

LA1

~~1.8.4 Not Used~~

1.8.5 Gaseous Radwaste Treatment System

A Gaseous Radwaste Treatment System is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

1.8.6 Ventilation Exhaust Treatment System

A Ventilation Exhaust Treatment System is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be Ventilation Exhaust Treatment System components.

LA1

1.8.7 Purge-Purging

Purge or Purging is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

A2

1.8.8 VENTING

~~Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during Venting. Vent, used in system names, does not imply a venting process.~~

1.8.9 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 its 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.~~

A2

1.8.10 UNRESTRICTED AREA

~~An Unrestricted Area shall be any area at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial institutional and/or recreational purposes.~~

1.9 CORE OPERATING LIMITS REPORT

CORE OPERATING LIMITS REPORT (COLR)

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification ~~6.9~~ 5.6.5. Plant operation within these core operating limits is addressed in individual specifications.

Add the Following definitions:

ACTIONS
ALLOWABLE THERMAL POWER
AXIAL POWER SHAPING RODS (APSRs)
CONTROL RODS
DOSE EQUIVALENT I-131
LEAKAGE
MODE
PHYSICS TESTS
SHUTDOWN MARGIN (SDM)
THERMAL POWER

A11

A1 <except as marked>

<See 3.4>

3.1.4 Reactor Coolant System Activity

Specification

The total activity of the reactor coolant due to nuclides with half lives longer than 30 minutes shall not exceed 224/E microcuries per ml whenever the reactor is critical. E is the average (mean) beta and gamma energies per disintegration, in MeV, weighted in proportion to the measured activity of the radionuclides in reactor coolant samples.

E - AVERAGE DISINTEGRATION ENERGY

Making up at least 95% of the non-iodine activity of the coolant

M4

Bases

The above specification is based on limiting the consequences of a postulated accident involving the double-ended rupture of a steam generator tube. The rupture of a steam generator tube enables reactor coolant and its associated activity to enter the secondary system where volatile isotopes could be discharged to the atmosphere through condenser air-ejectors and through steam safety valves (which may lift momentarily). Since the major portion of the activity entering the secondary system is due to noble gases, the bulk of the activity would be discharged to the atmosphere. The activity release continues until the operator stops the leakage by reducing the reactor coolant system pressure below the set point of the steam safety valves and isolates the faulty steam generator. The operator can identify the faulty steam generator by using the N^{16} detectors on the steam lines in conjunction with the off-gas monitors on the condenser air ejector lines; thus he can isolate the faulty steam generator within 34 minutes after the tube break occurred. During that 34 minute period, a maximum of 2760 ft^3 of 580°F reactor coolant leaked into the secondary system. (This is equivalent to a cold makeup volume of 1980 ft^3).

A26

The controlling dose of the steam generator tube rupture accident is the whole-body dose resulting from immersion in the cloud of released activity. To insure that the public is adequately protected, the specific activity of the reactor coolant will be limited to a value which will insure that the whole-body dose at the site boundary will not exceed 0.5 Rem should a steam generator tube rupture accident occur.

Although only volatile isotopes will be released from the secondary system, the following whole-body dose calculation conservatively assumes that all of the radioactivity which enters the secondary system with the reactor coolant is released to the atmosphere. Both the beta and gamma radiation from these isotopes contribute to the whole-body dose. The gamma dose is dependent on the finite size and configuration of the cloud. However the analysis employs the simple model of a semi-infinite cloud, which gives an upper limit to the potential gamma dose. The semi-infinite cloud model is applicable to the beta dose because of the short range of beta radiation in air. It is further assumed that meteorological conditions during the course of the accident correspond to Pasquill Type F and 1 meter per second wind speed, resulting in a X/Q value of 1.16 x 10^{-4} sec/m^3, which includes a correction factor of 2.2 to the dilution calculated by the Pasquill method.This correction factor was shown

Specification 1.2
1.3
1.4

<INSERT 1.2, Logical Connectors, as presented in the ITS.> A15

<INSERT 1.3, Completion Times, as presented in the ITS.> A16

<INSERT 1.4, Frequency, as presented in the ITS.> A17

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with NUREG-1430, Revision 1. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1430, Revision 1. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1430, Revision 1. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A2 CTS 1.4.6 provides a definition for Degree of Redundancy. CTS 1.8.1 provides a definition for Source Check. CTS 1.8.7 provides a definition for Purge-Purging. CTS 1.8.8 provides a definition for Venting. CTS 1.8.9 provides a definition for Member(s) of the Public. CTS 1.8.10 provides a definition for Unrestricted Area.

These definitions are not adopted in ITS because the CTS specification that use these definitions are not retained in the ITS; and the equivalent ITS specification does not use the defined term. The removal of a definition that is not used in the ONS ITS is an administrative change because it has no impact on the implementation of any existing requirement not addressed in the ONS ITS conversion. This change is consistent with the NUREG.

A3 CTS 1.1 defines Rated Power as a steady state reactor core output of 2568 MWt. ITS defines RATED THERMAL POWER as a total reactor core heat transfer rate to the reactor coolant of 2568 MWt. The NUREG definition for RATED THERMAL POWER is used in lieu of the ONS CTS definition of Rated Power. Although the CTS wording regarding steady state output is not retained, no technical or interpretational change exists. The maximum power level as prescribed in the facility operating license limits steady state output. Therefore, this change is administrative and is consistent with the NUREG.

A4 CTS 1.2 defines Reactor Operating Conditions. The CTS provides individual definitions for each Reactor Operating Condition. The ITS establishes MODES of operation which are comparable to the Reactor Operating Conditions defined in Section 1.2 of the CTS. The MODES comparable to these Conditions are defined by the combination of reactivity condition (K_{eff}), % 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 %Δk/k). The NUREG defines the reactivity condition in terms of K_{eff}. The ONS ITS adopts the K_{eff} convention. The small difference between 1 percent Δk/k and 0.99 K_{eff} is well within the typical accuracy for reactivity predictions. Therefore, the movement of the CTS definitions for Reactor Operating Conditions into the ITS Table 1.1-1 is considered an administrative change and is consistent with the NUREG method of presentation of MODES. The applicability of the Reactor Operating Condition definition changes are evaluated at each occurrence of the defined Reactor Operating Condition in the ONS CTS. Changes to the ONS CTS are discussed on an individual basis with the Specification. Each change is evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis.

A5 Not used.

A6 CTS 1.2.2 defines Hot Shutdown in terms of a subcritical condition (1% Δk/k shutdown) and an average reactor coolant temperature of greater than or equal to 525°F. This Hot Shutdown operating condition definition is modified to correlate with the ITS MODE 3 criteria established in ITS Table 1.1-1. The ITS MODE 3 criteria imposes a minimum average reactor coolant temperature of 250°F. The lower average reactor coolant temperature band could represent more restrictive requirements on the operation of the facility. The applicability of this Reactor Operating Condition definition change is evaluated at each occurrence of the defined Hot Shutdown Applicability in the CTS. Changes to the CTS are discussed on an individual basis with the Specification. Each change is evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis. This change is consistent with the NUREG.

A7 CTS 1.2.4 defines Hot Standby as when T_{avg} is greater than 525°F, the reactor is critical and indicated neutron power on the power range channels is less than 2 percent of rated power. ITS MODE 3 is defined as when $k_{eff} < 0.99$ and average reactor coolant temperature is $\geq$ 250°F. The ITS MODE 3 definition impose more stringent requirements on the facility. For example, ACTIONS in the CTS that presently direct the unit to Hot Standby (which allow critical operation at a power level below 2%) requires that the reactor be taken to a subcritical condition ($K_{eff} < 0.99$) in the ITS. Similarly, during a plant heatup, the ITS MODE 3 definition requires equipment to be placed into service at a lower operating temperature (250°F vice 525°F) than required by the CTS. The applicability of this Reactor Operating Condition definition change is evaluated at each occurrence of the defined Hot Standby Applicability in the ONS CTS. Changes to the CTS are discussed on an individual basis with the Specification. Each change is evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis. This change is an administrative change and is consistent with the NUREG.

A8 Not used.

A9 CTS 1.5.1 defines a Trip Test as a test of logic elements in a protective channel to verify their associated trip action. CTS 1.5.2 defines a Channel Test as the injection of an internal or external test signal into the channel to verify its proper output response; including alarm and/or trip initiating action where applicable. The ITS definition of CHANNEL FUNCTIONAL TEST requires 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. CTS 1.5.1 and 1.5.2 definitions for Channel Test and Trip Test, when combined, are comparable to the NUREG definition of CHANNEL FUNCTIONAL TEST. Therefore, the CHANNEL FUNCTIONAL TEST definition from the NUREG has been adopted in its entirety. This change is administrative and is consistent with the NUREG.

A10 Not used.

A11 CTS definitions comparable to the ITS definitions for ACTIONS, ALLOWABLE THERMAL POWER, AXIAL POWER SHAPING RODS (APSRS), CONTROL RODS, DOSE EQUIVALENT I-131, LEAKAGE, MODE, PHYSICS TEST, SHUTDOWN MARGIN and THERMAL POWER do not exist. ITS states ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times. ITS states ALLOWABLE THERMAL POWER shall be the maximum reactor core heat transfer rate to the reactor coolant permitted by consideration of the number and configuration of reactor coolant pumps (RCPs) in operation. ITS states APSRs shall be the control components with part length absorbers used to control the axial power distribution of the reactor core. The APSRs are positioned manually by the operator and do not trip. ITS states CONTROL RODS shall be all full length safety and regulating rods. ITS states 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 I-131 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."

ITS states 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;

   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 injection or 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.

ITS states 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.

ITS states 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 UFSAR;

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

ITS states 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.

ITS states THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

This change includes definitions in the ITS that are established in the NUREG but which do not exist as definitions in the CTS. The addition of the definitions is made to make the Improved Technical Specifications (ITS) consistent with the B&W Standard Technical Specification. 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. This change is consistent with the NUREG.

A12 Not used.

A13 CTS 1.2.4 and 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). CTS 1.2.8 does not include a requirement regarding power level. CTS 1.2.5 is comparable to ITS MODE 1 and CTS 1.2.4 combined with CTS 1.2.8 are comparable to ITS MODE 2. ITS MODE 1 and MODE 2 establishes the transition power level as 5% RATED THERMAL POWER in accordance with Table 1.1-1 of the NUREG. The 5% RTP MODE transition criteria is adopted for the purpose of maintaining consistency with the NUREG.

While the change in definition could be a less restrictive change, its affect cannot be adequately evaluated without considering how it is applied in each CTS occurrence. Therefore, the applicability of the Reactor Operating Condition definition changes are evaluated at each occurrence of the defined Reactor Operating Condition in the CTS. Changes to the CTS are discussed on an individual basis with the Specification. Each change is evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis. Where the overall affect was less or more restrictive an appropriate discussion is provided.

A14 The CTS 1.2.6 definition for Refueling Shutdown includes when, even with all rods removed, the reactor would be subcritical by at least 1 percent $\Delta k/k$ and the coolant temperature at the low pressure injection pump suction is no more than 140°F. ITS MODE 6 is when one or more reactor head bolts are not fully tensioned. This change results in the deletion from the definition of the requirement that the reactor be maintained subcritical by 1% $\Delta k/k$ 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). However, ITS LCO 3.9.1 provides controls upon SDM when in MODE 6. The adoption of ITS Specification 3.9.1 evaluates the implications of this change in definition and categorizes the adoption of ITS Specification 3.9.1 and its Bases as more restrictive or less restrictive as appropriate. This change is administrative and is consistent with the NUREG.

A15 CTS provisions comparable to ITS 1.2 do not exist. ITS 1.2 establishes the use and convention for the Logical Connectors used throughout the Improved Technical Specifications (ITS). In addition, ITS 1.2 demonstrates through example the usage of the Logical Connectors. This is an administrative change made to conform to the NUREG convention and is consistent with the NUREG. Technical changes associated with the adoption of these conventions are included in separate discussion of change associated with the individual specifications.

A16 CTS provisions comparable to ITS 1.3 do not exist. ITS 1.3 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. This is an administrative change made to conform to the NUREG convention and is consistent with the NUREG. Technical changes associated with the adoption of these conventions are included in separate discussion of change associated with the individual specifications.

A17 CTS provisions comparable to ITS 1.4 do not exist. ITS 1.4 establishes the use and convention of 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. This is an administrative change made to conform to the NUREG convention and is consistent with the NUREG. Technical changes associated with the adoption of these conventions are included in separate discussion of change associated with the individual specifications.

A18 CTS 1.2.1, 1.2.2 and 1.2.4 use the term T_{avg}. ITS Table 1.1-1 uses the term "Average Reactor Coolant Temperature." No technical or interpretational change exists. This change is administrative and is consistent with the NUREG.

A19 CTS 1.2.4 and 1.2.5 provide definitions comparable to the ITS requirements defining MODE 2 and MODE 1 respectively. These CTS definitions specify power limits in terms of indicated neutron power. ITS Table 1.1-1 prescribes power limits in terms % RATED THERMAL POWER. Indicated neutron power is normalized to calorimetric values. Therefore, indicated neutron power is equivalent to % RATED THERMAL

POWER. Therefore this change is administrative and is consistent with the NUREG.

A20 A CTS definition comparable to ITS MODE 4 does not exist. ITS defines MODE 4 as K_{eff} < 0.99 K_{eff} and average reactor coolant temperature < 250°F and > 200°F. When in this condition, CTS has no defined Reactor Operating Condition. This change is consistent with the NUREG presentation of MODES. The applicability of the Reactor Operating Condition definition changes are evaluated at each occurrence of the defined Reactor Operating Condition in the CTS. Changes to the CTS are discussed on an individual basis with the Specification. Each change is evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis. This change administrative and is consistent with the NUREG.

A21 A provision comparable to last sentence to the ITS definition of CORE ALTERATION does not exist. This sentence states that suspension of CORE ALTERATIONS does not preclude completion of movement of a component to a safe position. The is no CTS requirement directing the suspension of Refueling Operations. Therefore, no relief is provided by the addition of this statement. This change is an administrative change and is consistent with the NUREG.

A22 Not used.

A23 CTS 1.5.4 does not include an explicit provision for including displays in the Instrument Channel Calibration but does require the calibration to encompass the entire channel. ITS explicitly includes displays within the definition of CHANNEL CALIBRATION. Since CTS 1.5.4 requires the Instrument Channel Calibration to encompass the entire channel and a display is part of a channel, this change is an administrative change and is consistent with the NUREG.

A24 An explicit CTS provision permitting Instrument Channel Calibration to be performed by any sequential, overlapping or total channel steps does not exist. The ITS definition of CHANNEL CALIBRATION explicitly permits the testing to be performed by any sequential, overlapping or total channel steps so the entire channel is calibrated. CTS does not preclude such testing using any sequential, overlapping or total channel steps so the entire channel is calibrated. Therefore, this change is an administrative change and is consistent with the NUREG.

A25 CTS 1.2.1 and 1.2.6 contain a reference to a separate CTS specification which provides requirements regarding RCS pressure. CTS 1.6.2 provides a information regarding a description of neutron power range channel inputs into imbalance instrumentation and information regarding a reference to a separate CTS specification which contains imbalance limits and setpoints. The description and references are not retained in the ITS. No technical or interpretational change exists. This change is administrative and is consistent with the NUREG.

A26 The ONS 1, 2, & 3 CTS Bases are completely replaced by revised bases that reflect the format and applicable content of proposed ITS Section 3.4. The revised Bases are shown in the proposed ONS ITS Bases for Section 3.4.

A27 An explicit CTS provision permitting Instrument Channel Testing to be performed by any sequential, overlapping or total channel steps does not exist. The ITS definition of CHANNEL FUNCTIONAL TEST explicitly permits the testing to be performed by any sequential, overlapping or total channel steps so the entire channel is functionally tested. CTS does not preclude such testing using any sequential, overlapping or total channel steps so the entire channel is calibrated. Therefore, this change is an administrative change and is consistent with the NUREG.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 1.2.6 defines Refueling Shutdown as when, even with all rods removed, the reactor would be subcritical by at least 1 percent $\Delta k/k$ and the coolant temperature at the low pressure injection pump suction is no more than 140°F. ITS 1.1 incorporates the ISTS definition for MODE 6 - Refueling, including footnote (c) in ITS Table 1.1-1, "One or more reactor vessel head closure bolts less than fully tensioned." CTS provisions comparable to ITS Table 1.1-1, Note B do not exist. ITS Table 1.1-1, Note (b) is complementary to Note (c) and serves to fully differentiate ITS MODE 4 and MODE 5 from MODE 6.

Since MODE 6 is entered whenever the first reactor head closure bolt is de-tensioned regardless of low pressure injection pump suction temperature, this change is a more restrictive requirement and is consistent with the NUREG. This change is acceptable since it has no significant impact on plant operations while serving to more clearly define the unit's transition into or out of MODE 6.

M2 CTS 1.2.7 defines refueling operation as a change in core geometry by manipulation of fuel or control rods when the head is removed and fuel is in the vessel. The ITS definition of CORE ALTERATION is comparable to CTS 1.2.7 but also includes movement of sources within the reactor vessel. The inclusion of source movement within the reactor vessel is a more restrictive requirement upon unit operation and is consistent with the NUREG. The inclusion of source movement within the reactor vessel as a core alteration is appropriate since source movement can affect core criticality.

M3 The CTS 1.6.4 definition for Channel Functional Test does not specify the point of test signal injection. The ITS definition for CHANNEL FUNCTIONAL TEST requires, " . . . injection of a signal into the channel as close to the sensor as practicable . . . ." The additional requirement regarding the point of signal injection is a more restrictive requirement upon unit operation and is consistent with the NUREG. Injection of the signal as close as practicable to the sensor is acceptable since it ensures a more comprehensive test of the complete instrument channel.

M4 CTS 3.1.4 defines $\bar{E}$ as the average (mean) beta and gamma energies per disintegration, in MeV, weighted in proportion to the measured activity of the radionuclides in reactor coolant samples. The ITS definition for E as the average mean (weighted in proportion to the measured activity of each radionuclide in the reactor coolant at the time of sampling) beta and gamma energies per disintegration (in MeV) for isotopes, with half lives > 30 minutes, making up at least 95% of the total noniodine activity in the coolant. The added specificity regarding isotopic composition (at least 95% of the total noniodine activity in the coolant) is a more restrictive requirement upon unit operation and is consistent with the NUREG.

TECHNICAL CHANGE - REMOVAL OF DETAILS

LA1 CTS 1.8.3 specifies requirements related to the Process Control Program. CTS 1.8.5 and 1.8.6 provide a definition for the Gaseous Radwaste Treatment System and Ventilation Exhaust Treatment System respectively. These requirements are relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since 10 CFR 20, 10 CFR 61, 10 CFR 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste and Radioactive Gaseous Effluents still require overall compliance with applicable shipping, burial and effluent release requirements. These relocated requirements are duplicative/contained in other regulations or are required to comply with regulations. UFSAR requirements are not allowed to be changed without 10 CFR 50.59 evaluation, which ensures any changes are appropriately reviewed. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of this detail is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. The relocation of this RETS requirements is consistent with Generic Letter 89-01 with the exception of relocation to the UFSAR Chapter 16 instead of the ODCM and PCP. The NRC found relocation to the UFSAR was an acceptable alternative to relocation to the ODCM and PCP in the SER issued on 1/22/91 for license amendments 187, 187, 184 for Units 1, 2 and 3 respectively issued.

LA2 CTS 1.4.1, 1.4.2, 1.4.3, 1.4.4 and 1.4.5 provide descriptive information regarding instrumentation. CTS 1.5.5 and 1.5.6 provide descriptive information regarding a Heat Balance Check and a Heat Balance Calibration respectively. This information is relocated to the ITS Bases. This information is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for system OPERABILITY. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. Changes to the Bases are controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

TECHNICAL CHANGES - MORE RESTRICTIVE

L1 CTS 1.5.2 defines a Channel Test as the injection of an internal or external test signal into the channel to verify its proper output response; including alarm and/or trip initiating action where applicable. The ITS Defines a CHANNEL FUNCTIONAL TEST as 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. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the channel is functionally tested. The provision to permit use an actual signal is a less restrictive requirement upon unit operation and is consistent with the NUREG. Use of an actual signal does not affect the performance of the channel. OPERABILITY can be adequately demonstrated in either case since the channel itself does not discriminate between an "actual" or "simulated" signal.

L2 CTS 1.5.4 requires an Instrument Channel Calibration to encompass the entire channel and does not exclude RTDs and thermocouples. The ITS definition of CHANNEL CALIBRATION allows performing "...an in place qualitative assessment of sensor behavior..." for these devices. This change is a less restrictive requirement upon unit operations and is consistent with the NUREG. A qualitative assessment of sensor behavior is acceptable for RTDs and thermocouples since the operation of these devices is governed by well understood and predictable physical relationships between the temperature of the sensed medium and the output of the RTD or thermocouple. Additionally, the output of RTDs and thermocouples is not adjustable. These devices are reliable and not subject to drift in the same manner as other sensors. As a result a qualitative assessment of sensor behavior is sufficient to determine its OPERABILITY and acceptability for continued use.

L3 CTS 1.2.8 defines Startup as a reduction in the shutdown margin with the intent of going critical. ITS MODE 2 is comparable to the CTS condition of startup. ITS MODE 2 is specified as when $K_{eff} \geq 0.99$ and THERMAL POWER is < 5%. The elimination of requirements associated with the CTS Startup Condition when shutdown margin is being reduced and $K_{eff} < 0.99$ (i.e., when in ITS MODES 3, 4 and 5) is a less restrictive requirement upon unit operation and is consistent with the NUREG. When the unit is in MODES 3, 4 or 5 shutdown margin is controlled by ITS Specification 3.4.1. ITS 3.4.1 ensures appropriate control upon shutdown margin when in ITS MODES 3, 4 or 5.

L4 CTS 1.2.4 and 1.2.5 require use of the power range neutron channels as the indication to be used to establish the transition between Hot Standby and Power Operation. ITS Table 1.1-1 uses % RATED THERMAL POWER as the measure of reactor power used to establish the transition point between MODES 1 and 2. Although, the power range neutron channels may still be used since they are normalized to the calorimetric power measurement, the ITS permits use of the calorimetric measurement itself. The added flexibility to use calorimetric indications as a measure of THERMAL POWER is a less restrictive requirement upon unit operation and is consistent with the NUREG. The use of the calorimetric indication is acceptable since it provides a more direct and more accurate indication of THERMAL POWER.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 1.0 - USE AND APPLICATION

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS

ADMINISTRATIVE CHANGES

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." Some of the proposed changes involve reformatting, renumbering, and rewording of Technical Specifications. These changes, since they do not involve technical changes to the Technical Specifications, are administrative.

This type of change is connected with the movement of requirements within the current requirements, or with the modification of wording which does not affect the technical content of the current Technical Specifications. These changes will also include nontechnical modifications of requirements to conform to the Writer's Guide or provide consistency with the Improved Standard Technical Specifications in NUREG-1430. Administrative changes are not intended to add, delete, or relocate any technical requirements of the current Technical Specifications.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed changes involve reformatting, renumbering, and rewording of the existing Technical Specification. These modifications involve no technical changes to the existing Technical Specifications. The majority of changes were done in order to be consistent with NUREG-1430. During the development of NUREG-1430, certain wording preferences or English language conventions were adopted. The changes are administrative in nature and do not impact initiators of analyzed events. They also do not impact the assumed mitigation of accidents or transient events. Therefore, the changes do 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 changes involve reformatting, renumbering, and rewording of the existing Technical Specifications. The changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The changes will not impose any new or different requirements or eliminate any existing

requirements. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes involve reformatting, renumbering, and rewording of the existing Technical Specifications. The changes are administrative in nature and will not involve any technical changes. The changes will not reduce a margin of safety because it has no impact on any safety analysis assumptions. Also, since these changes are administrative in nature, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGES - REMOVAL OF DETAILS

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." Some of the proposed changes involve moving details (engineering, procedural, etc.) out of the Technical Specifications and into a licensee controlled document. This information may be moved to the ITS Bases, UFSAR, or other programs controlled by the licensee. The removal of this information is considered to be less restrictive because it is no longer controlled by the Technical Specification change process. Typically, the information moved is descriptive in nature and its removal conforms with NUREG-1430 for format and content.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed changes move details from the Technical Specifications to a licensee controlled document. The changes do not result in any hardware or operating procedure changes. The details being removed from the Technical Specifications are not assumed to be an initiator of any analyzed event. The licensee controlled documents containing the removed Technical Specification details are maintained using the provisions of 10 CFR 50.59, 10 CFR 50.54(a), 10 CFR 50.55(a), or other established review and control programs. Since changes to a licensee controlled document are evaluated per 10 CFR 50.59, 10 CFR 50.54(a), 10 CFR 50.55(a), or other established review and control programs, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated is involved. Therefore, the changes do 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 changes move detail from the Technical Specifications to a licensee controlled document. The changes will not alter the plant configuration (no new or different type of equipment will be installed) or make changes in methods governing normal plant operation. The changes will not impose different requirements, and adequate control of information will be maintained. The changes will not alter assumptions made in the safety analysis and licensing basis. Therefore, the changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes move detail from Technical Specifications to a licensee controlled document. The changes do not reduce the margin of safety since the location of details has no impact on any safety analysis assumptions. In addition, the details to be transposed from the Technical Specification to a licensee controlled document are the same as the existing Technical Specification. Future changes to this licensee controlled document will be evaluated per the requirements of 10 CFR 50.59, 10 CFR 50.54(a), 10 CFR 50.55(a), or other established review and control programs.

LESS RESTRICTIVE CHANGE L1

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 1.5.2 defines a Channel Test as the injection of an internal or external test signal into the channel to verify its proper output response; including alarm and/or trip initiating action where applicable. The ITS Defines a CHANNEL FUNCTIONAL TEST as 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. The provision to permit use an actual signal is a less restrictive requirement upon unit operation and is consistent with the NUREG. Use of an actual signal does not affect the performance of the channel. OPERABILITY can be adequately demonstrated in either case since the channel itself does not discriminate between an "actual" or "simulated" signal.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change allows use of an actual signal to be used in the CHANNEL FUNCTIONAL TEST. This change does not result in any hardware changes. The CHANNEL FUNCTIONAL TEST is not considered as the initiator of any previously analyzed accident. As such, the probability of an accident is independent of the signal used to perform CHANNEL FUNCTIONAL TESTING. Also, the change does not change the assumed response of the equipment in performing its specified mitigation functions from that originally considered. The consequences are not changed since the instrument channel functions the same, regardless of the signal used to perform the CHANNEL FUNCTIONAL TEST. Therefore, the change does not significantly increase the probability or consequences of an 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 still ensure proper availability for the required instrument functions. 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?**

This change to permit the use of an actual signal in addition to a test signal does not involve a change in setpoints and cannot affect any margin of safety associated with the response to a design basis accident. The instrumentation channel functions the same regardless of the source of the initiation signal. Therefore, this change to permit use of an actual signal is not considered to involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L2

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 1.5.4 requires an Instrument Channel Calibration to encompass the entire channel and does not exclude RTDs and thermocouples. The ITS definition of CHANNEL CALIBRATION allows performing "...an in place qualitative assessment of sensor behavior..." for these devices. This change is a less restrictive requirement upon unit operations and is consistent with the NUREG. A qualitative assessment of sensor behavior is acceptable for RTDs and thermocouples since the operation of these devices is governed by well understood and predictable physical relationships between the temperature of the sensed medium and the output of the RTD or thermocouple. Additionally, the output of RTDs and thermocouples is not adjustable. These devices are reliable and not subject to drift in the same manner as other sensors. As a result a qualitative assessment of sensor behavior is sufficient to determine its OPERABILITY and acceptability for continued use.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change allows permits calibration of RTDS and thermocouples to consist of an in place qualitative assessment of sensor behavior. This change does not result in any hardware changes. The RTDs and thermocouples are not an initiator of any previously analyzed accident. As such, the probability of an accident is independent of the manner of calibration. Also, the change does not change the assumed response of the equipment in performing its specified mitigation functions from that originally considered. The consequences are not changed since the instrument channels functions the same, regardless of the calibration methodology. Therefore, the change does not significantly increase the probability or consequences of an 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 still ensures proper availability for the required instrument functions. 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?**

This change to the calibration requirements for RTDs and thermocouples does not involve a change in setpoints and does not affect any margin of safety associated with the response to a design basis accident. Therefore, this change to permit calibration of RTDs and thermocouples to consist of an in place qualitative assessment of sensor behavior is not considered to involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L3

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 1.2.8 defines Startup as a reduction in the shutdown margin with the intent of going critical. ITS MODE 2 is comparable to the CTS condition of startup. ITS MODE 2 is specified as when $K_{eff} \geq 0.99$ and THERMAL POWER is < 5%. The elimination of requirements associated with the CTS Startup Condition when shutdown margin is being reduced and $K_{eff} < 0.99$ (i.e., when in ITS MODES 3, 4 and 5) is a less restrictive requirement upon unit operation and is consistent with the NUREG. When the unit is in MODES 3, 4 or 5 shutdown margin is controlled by ITS Specification 3.4.1. ITS 3.4.1 ensures appropriate control upon shutdown margin when in ITS MODES 3, 4 or 5.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change allows operation when K_{eff} is < 0.99 without restrictions associated with being in the Startup condition. This change does not result in any hardware changes. Restrictions associated with being in the Startup condition when K_{eff} is < 0.99 are not considered as initiators of any previously analyzed accident. As such, the probability of an accident is independent of being in the Startup condition when K_{eff} is < 0.99. Also, the change does not change the assumed response of equipment in performing specified mitigation functions from that originally considered. The consequences are not changed since the equipment functions the same. Therefore, the change does not significantly increase the probability or consequences of an 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 still ensures proper availability for the required equipment functions. 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?**

   This change to the requirements does not involve a change in setpoints and does not affect any margin of safety associated with the response to a design basis accident. Therefore, this change does not involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L4

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 1.2.4 and 1.2.5 require use of the power range neutron channels as the indication to be used to establish the transition between Hot Standby and Power Operation. ITS Table 1.1-1 uses % RATED THERMAL POWER as the measure of reactor power used to establish the transition point between MODES 1 and 2. Although, the power range neutron channels may still be used since they are normalized to the calorimetric power measurement, the ITS permits use of the calorimetric measurement itself. The added flexibility to use calorimetric indications as a measure of THERMAL POWER is a less restrictive requirement upon unit operation and is consistent with the NUREG. The use of the calorimetric indication is acceptable since it provides a more direct and more accurate indication of THERMAL POWER.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change permits use of alternate indications of reactor THERMAL POWER. This change does not result in any hardware changes. The indication used to indicate THERMAL POWER is not considered as the initiator of any previously analyzed accident. As such, the probability of an accident is independent of the indication of THERMAL POWER used. Also, the change does not change the assumed response of equipment in performing specified mitigation functions from that originally considered. The consequences are not changed since the instruments functions the same. Therefore, the change does not significantly increase the probability or consequences of an 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 still ensure proper availability for the required functions. 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?**

   This use of acceptable alternatives for the indication of THERMAL POWER cannot affect any margin of safety associated with the response to a design basis accident. Therefore, this change to allow the used of alternative indications for THERMAL POWER, is not considered to involve a significant reduction in the margin of safety.

ENVIRONMENTAL ASSESSMENT

This proposed Technical Specification Change has been evaluated against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed changes meet the criteria for categorical exclusion as provided for under 10 CFR 51.22 (c) (9). The following is a discussion of how the proposed Technical Specification Change meets the criteria for categorical exclusion.

10 CFR 51.22 (c) (9): Although the proposed change involves changes to requirements with respect to inspection or surveillance requirements;

(i) the proposed change involves no Significant Hazards Consideration (refer to the No Significant Hazards Consideration section of this Technical Specification Change Request),

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite since the proposed changes do not affect the generation of any radioactive effluents nor do they affect any of the permitted release paths, and

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Based on the aforementioned and pursuant to 10 CFR 51.22 (b), no environmental assessment or environmental impact statement need be prepared in connection with issuance of an amendment to the Technical Specifications incorporating the proposed changes of this request.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 1.0 - USE AND APPLICATION

ATTACHMENT 5

NUREG 1430 MARKUP AND JUSTIFICATIONS

TECHNICAL SPECIFICATIONS

1.0 USE AND APPLICATION

CTS

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.	Doc A11
ALLOWABLE THERMAL POWER	ALLOWABLE THERMAL POWER shall be the maximum reactor core heat transfer rate to the reactor coolant permitted by consideration of the number and configuration of reactor coolant pumps (RCPs) in operation.	Doc A11
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 5
AXIAL POWER SHAPING RODS (APSRs)	APSRs shall be control components with part length absorbers used to control the axial power distribution of the reactor core. The APSRs are positioned manually by the operator and are not trippable.	Doc A11
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. 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~~	1.5.4 TSTF 19, R1

(continued)

1.1 Definitions

TSTF 19, R1

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.~~ 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. 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.~~ 7

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. 1.5.1 1.5.2

~~The ESFAS CHANNEL FUNCTIONAL TEST shall also include testing of ESFAS safety related bypass functions for each channel affected by bypass operation.~~ 8

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the channel is functionally tested

Doc A11

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~~. 9

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

TSTF 39, R1

(continued)

1.1 Definitions

CORE ALTERATION (continued)	ALTERATIONS shall not preclude completion of movement of a component to a safe position.	1.2.7
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.	1.9
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"]~~.	DOC A11 23
$\bar{E}$ – 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]~~ 30 minutes, making up at least 95% of the total noniodine activity in the coolant.	3.1.4 11
~~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.~~	TSTF 125
~~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~~	13

(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.

13

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.

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).

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LEAKAGE

LEAKAGE shall be:

Doc A11

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;

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)

1.1 Definitions

LEAKAGE (continued)

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

b. Unidentified LEAKAGE

All LEAKAGE except RCP seal water injection or leakoff that is not identified LEAKAGE ~~or controlled LEAKAGE~~. TSTF 40

c. Pressure Boundary LEAKAGE

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. DOC All

~~**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.~~ 25

~~**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)

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. Doc A 11

These tests are:

a. Described in ~~Chapter [14, Initial Test Program] of~~ the FSAR; 19 U

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

~~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."~~ 20

QUADRANT POWER TILT (QPT)

QPT shall be defined by the following equation and is expressed as a percentage. 1.6.1

$$QPT = 100 \left(\frac{\text{Power in any Core Quadrant}}{\text{Average Power of all Quadrants}} - 1 \right)$$

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~[2544]~~ 2568 MWt. 1.1 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.~~ 13

(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:	Doc A11
	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	1
	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.	1.5.7
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.	Doc A11

 CTS

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)	26
1	Power Operation	≥ 0.99	> 5	NA	1.2.5
2	Startup	≥ 0.99	≤ 5	NA	1.2.3 1.2.4
3	Hot Standby	< 0.99	NA	≥ ~~[330]~~ 250	1.2.2
4	Hot Shutdown[b]	< 0.99	NA	~~[330]~~ 250 > T_{avg} > ~~[~~200~~]~~	Doc A20
5	Cold Shutdown[b]	< 0.99	NA	≤ ~~[~~200~~]~~ 1	1.2.1
6	Refueling[c]	NA	NA	NA	1.2.6

24

(a) Excluding decay heat. Doc A19

(b) All reactor vessel head closure bolts fully tensioned. Doc M1

(c) One or more reactor vessel head closure bolts less than fully tensioned. Doc M1

Doc A15 for all of 1.2

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.

Doc A16 for all of 1.3 except as marked

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 not 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.

(continued)

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)

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 24 21 ~~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.	6 hours
	AND	
	B.2 Be in MODE 4.	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. <u>AND</u> 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)

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 22
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

(continued)

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

Doc A17 for all of 1.4

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.

TECHNICAL SPECIFICATIONS

NOTE: The first four justifications for these changes from NUREG-1430 were generically used throughout the individual LCO section markups. Not all generic justifications are used in each section.

1. The brackets are removed and the proper plant specific information or value is provided.

2. Not used.

3. Not used.

4. Not used.

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

6 Not used.

7 The last paragraph of the definition for CHANNEL CALIBRATION is modified to eliminate the references to safety related bypasses for the Reactor Protective System (RPS), Engineered Safety Feature Actuation System (ESFAS) and the Emergency Feedwater Initiation and Control (EFIC). The sentence is deleted since it imposes testing requirements of bypass features in excess of the requirements of the CTS. The CTS Specifications that require the CHANNEL CALIBRATION to be applied to bypass functions are retained in the ITS.

8 The last sentence of the definition for CHANNEL FUNCTIONAL TEST is not adopted. The ONS design does not include bypasses having a safety function in the ESFAS.

9 The definition of CONTROL RODS is modified to eliminate the overly prescriptive description of the function of the CONTROL RODS. The definition as written, if literally interpreted, prevents these reactivity control components from being used to startup the reactor, control xenon oscillations and control reactor imbalance, etc. Additionally, the NUREG and ITS usage of the term CONTROL RODS is

intended to classify a component for the purposes of application of LCOs and SRs and not to prescribe the function of the devices. [ONS-005]

10 Not used.

11 The CLB regarding the half lives of nuclides included in the E-bar determination is retained. The definition is changed from including nuclides with half lives longer than 15 minutes to including nuclides with half lives longer than 30 minutes. The CLB definition is retained in order to maintain consistency in the approach to determining offsite doses for certain accident analyses.

12 Not used.

13 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 are not consistent with CTS. Response time testing of these systems, as required by specifications in NUREG 1430, is not required by CTS Specifications.

14 Not used.

15 Not used.

16 Not used.

17 Not used.

18 Not used.

19 The specific chapter reference in part "a." of the PHYSICS TESTS definition is deleted. ONS is not a Standard Review Plant. Consequently Physic Testing is described in more than one UFSAR Chapter. Removal of the reference to a specific chapter simply ensures that physics testing described in other Chapters of the UFSAR is encompassed by this definition.

20 ONS 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.

21 The COMPLETION TIME in EXAMPLE 1.3-3 for REQUIRED ACTION C.1 and C.2 is changed from 72 hours to 24 hours. This change is made to provide a more representative example of this situation and to avoid possible confusion from the use of the same COMPLETION TIME in both CONDITION B and CONDITION C. [ONS-006]

22 The REQUIRED ACTION A.2 in Example 1.3-6 is changed from "Reduce THERMAL POWER to ≤ 50% RTP" to "Place the channel in bypass." This change is made to provide a REQUIRED ACTION in A.2 which is not automatically accomplished by performing the REQUIRED ACTION in B.1. This is done to provide a more representative and useful example. [ONS-006]

23 The second reference provided in the NUREG definition for Dose Equivalent I-131 is deleted. UFSAR 15.14.7 uses the dose conversion factors specified in TID-14844 for the determination of Dose Equivalent I-131.

24 The subscript AVG is deleted to preclude in any potential confusion with Oconee instrumentation that measures reactor coolant temperature. Oconee has a T_{AVG} indication. However, this instrument cannot be used to measure reactor coolant temperature in the range specified in ITS Table 1.1-1.

25 The NUREG Definitions for NUCLEAR HEAT FLUX HOT CHANNEL FACTOR $F_Q(Z)$ and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F_{\Delta H}^{N}$) are not adopted since the Specification, 3.2.5, "Power Peaking Factors" which uses these defined terms is not adopted.

26 An appropriate value of 250°F is selected for the MODE 4/MODE 3 transition temperature. There is no CTS equivalent value since the CTS does not include a similar MODE transition point. This temperature is appropriate since it is only slightly above the upper limit for using the Low Pressure Injection System in decay heat removal (DHR) alignment. This value permits use of MODE 4 as a transition MODE wherein required RCS flow may be maintained using either the RCS loops or the DHR loops.

27 The definition for L_a is not adopted in the ITS. The implementation of 10 CFR 50 Appendix J, Option B has resulted in modifications to the ISTS which capture the Leakage Rate Testing Program requirements in ITS paragraph 5.5.2. ITS section 5.5.16 provides a description of L_a which is consistent with the definition of L_a in ISTS Section 1.1, Definitions.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 1.0 - USE AND APPLICATION

ATTACHMENT 6

NUREG 1430 MARKUP AND JUSTIFICATIONS

BASES

There are no bases associated with ITS Section 1.0.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 2.0 - SAFETY LIMITS

ATTACHMENT 1

TECHNICAL SPECIFICATIONS

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.

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

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 2.1.2 is violated, restore compliance within limits and be in MODE 3 within 1 hour.

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

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 2.0 - SAFETY LIMITS

ATTACHMENT 2

BASES

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

ONS Design Criteria (Ref. 1) require that reactor core SLs ensure specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated transients. 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.

DNB is not a directly measurable parameter during operation, but neutron power and Reactor Coolant System (RCS) temperature, flow and pressure can be related to DNB using a critical heat flux (CHF) correlation. The BWC (Ref. 2) CHF correlation has been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BWC correlation applies to Mark-BZ fuel. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual local 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 1.18 (BWC).

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. 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

(continued)

BASES

BACKGROUND (continued)

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 to 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) and main steam relief valves (MSRVs) prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and anticipated transients. 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 (Ref. 3), in combination with all the LCOs, are designed to prevent any analyzed combination of transient conditions for RCS temperature, flow and pressure, and THERMAL POWER level that would result in a DNB ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

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

a. RCS High Pressure trip;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. RCS Low Pressure trip;

c. Nuclear Overpower trip;

d. RCS Variable Low Pressure trip;

e. Reactor Coolant Pump to Power trip;

f. Flux/Flow Imbalance trip;

g. High Core Outlet Temperature trip; and

h. MSRVs.

The SL represents a design requirement for establishing the RPS trip setpoints identified previously.

SAFETY LIMITS

SL 2.1.1.1 and SL 2.1.1.2 ensure that fuel centerline temperature stays below the melting point and that the minimum DNBR is not less than the safety analyses limit.

The SLs are preserved by monitoring process variables, AXIAL POWER IMBALANCE and Variable Low RCS Pressure, to ensure that the core operates within the fuel design criteria. AXIAL POWER IMBALANCE and Variable Low RCS Pressure protective limits are provided in the COLR. The trip setpoints are derived by adjusting the measurement system independent AXIAL POWER IMBALANCE and Variable Low RCS Pressure protective limits given in the COLR to allow for measurement system observability and instrumentation errors.

Operation within these limits is ensured by compliance with the AXIAL POWER IMBALANCE and Variable Low RCS Pressure protective limits preserved by their corresponding RPS setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," as specified in the COLR. The AXIAL POWER IMBALANCE protective limits are separate and distinct from the AXIAL POWER IMBALANCE operating limits defined by LCO 3.2.2, "AXIAL POWER IMBALANCE Operating Limits." The AXIAL POWER IMBALANCE operating limits in LCO 3.2.2, also specified in the COLR, preserve initial conditions of the safety analyses but are not reactor core SLs.

(continued)

BASES (continued)

APPLICABILITY

SL 2.1.1.1 and SL 2.1.1.2 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 MSRVs, 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.

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

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the requirement to go to MODE 3 places the unit in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

REFERENCES

1. UFSAR, Section 3.1.
2. BAW-10143P, Part 2, "Correlation of 15x15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation," August 1981.
3. UFSAR, Chapter 15.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

According to ONS Design Criteria (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation nor during anticipated transients. ONS Design Criteria (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 anticipated transients, 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). Hence, the safety limit is 2750 psig. To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure prior to initial operation, according to the ASME Code requirements. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

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).

The limiting peak pressure transient, as determined by the safety analyses (Ref. 5), is performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

a. Pressurizer power operated relief valve (PORV);

b. Steam line turbine bypass valves;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 (Ref. 2), is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section 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 and steam generators, 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.

2.2.2

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

(continued)

BASES

SAFETY LIMIT VIOLATIONS

2.2.2 (continued)

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 of operation where the potential for challenges to safety systems is minimized.

2.2.3

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.

REFERENCES

1. UFSAR, Section 3.1.
2. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
4. ASME USAS B31.7, Nuclear Power Piping, dated February 1968 with June 1968 Errata.
5. UFSAR, Chapters 5 and 15.
6. 10 CFR 100.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 2.0 - SAFETY LIMITS

ATTACHMENT 3

CTS MARKUP AND DISCUSSION OF CHANGES

A1 <Except as marked>

2.0 ~~SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS~~ SAFETY LIMITS (SLs)

2.1 ~~SAFETY LIMITS,~~ REACTOR CORE SLs

Applicability

Applic MODES 1 + 2

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

In MODES 1 + 2 — M1

Objective

To maintain the integrity of the fuel cladding.

Specification

≤ — A3

2.1.1.1 The maximum local fuel pin centerline temperature shall be less than 5080 - $(6.5\times10^{-3})\times$(Burnup, MWD/MTU) °F. Operation within this limit is assured by compliance with the Axial Power Imbalance Protective Limits as specified in the Core Operating Limits Report. — LA1

2.1.1.2 The DNBR shall be maintained greater than the correlation limits of ~~1.3 for BAW-2 and~~ 1.18 for BWC. — A4 Operation within this limit is assured by compliance with the Axial Power Imbalance Protective Limits and Variable Low RCS Pressure Protective Limits as specified in the Core Operating Limits Report. — LA1

Bases — A2

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 and anticipated transients. This is accomplished by operating within the nuclear boiling heat transfer regime where the heat transfer coefficient is large and the cladding temperature is only slightly greater than the coolant temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation, but neutron power and reactor coolant pressure and temperature can be related to DNB using a critical heat flux (CHF) correlation. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual local heat flux, is indicative of the margin to DNB.

The BAW-2 and BWC CHF correlations[1,2] have been developed to predict 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 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). 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.

A1 <except as marked>

1.

2.2 ~~SAFETY LIMITS -~~ ~~REACTOR COOLANT SYSTEM~~ PRESSURE SL

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.

Specification

L1 In MODES 1, 2, 3, 4, + 5

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig ~~when there are fuel assemblies in the reactor vessel.~~

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 1967.

<See Markup for ITS 3.4>

Bases

A2

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% of design pressure. [2] The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under USAS Section B31.7 is 110% of design pressure. Thus, the safety limit of 2750 psig (110% of the 2500 psig design pressure) has been established. [3] The settings, the reactor high pressure trip (2355 psig) and the pressurizer safety valves (2500 psig) [4] have been established to assure never reaching the reactor coolant system pressure safety limit. The initial hydrostatic test was conducted at 3125 psig (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the Reactor Coolant pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2450 psig.

REFERENCES

(1) FSAR, Section 5

(2) FSAR, Section 5.2.3.10.1

(3) FSAR, Section 5.2.2.3, Table 5.4-7

(4) FSAR, Section 5.4.6, Table 5.1-1

M2 A1 <except as marked> Specification 2.0

restore to within limits and

in MODE 3 within 1 hour

L3 L3

2.2 ~~6.3~~ ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED

2.2.1 ~~6.3.1~~ If a safety limit is exceeded, the reactor shall be shut down immediately and maintained in a safe shutdown condition until the Commission authorizes resumption of operation.

2.2.2

A5

6.3.2 The violation of a safety limit shall be reported to the Commission the Site Vice President and the Director of the Nuclear Safety Review Board.

LA2

6.3.3 A report shall be prepared which describes (1) applicable circumstances preceding the violation, (2) effects of the violation upon structures, systems, or components, and (3) corrective action taken to prevent recurrence. The report shall be reviewed by the Operations Superintendent and the Station Manager. The report shall be submitted to the Site Vice President and the Director of the Nuclear Safety Review Board.

6.3.4 A report of the violation, with appropriate analyses and corrective action to prevent recurrence shall be submitted to the Commission within 10 days of the violation.

L2

Add 2.2.3 — M3

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with NUREG-1430, Revision 1. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1430, Revision 1. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1430, Revision 1. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A2 Current technical specification (CTS) Bases will be administratively deleted in their entirety in favor of the NUREG Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 CTS 2.1 requires the maximum local fuel pin centerline temperature to be less than 5080 - (6.5×10^{-3}) x (Burnup, MWD/MTU)°F. ITS 2.1.1.1 requires the maximum local fuel pin centerline temperature to be $\leq$ 5080 - (6.5×10^{-3}) x (Burnup, MWD/MTU)°F. This minor difference (i.e., < versus $\leq$) is so close as to be imperceptible and is therefore considered administrative. The proposed changes are consistent with the NUREG.

A4 CTS 2.1 currently specifies that the DNBR shall be maintained greater than the correlation limits of 1.3 for BAW-2 and 1.18 for BWC. Since the BAW-2 correlation is no longer applicable (Oconee no longer uses Mark B fuel) it is not included in the ITS. As such, the deletion is considered administrative and is consistent with the NUREG.

A5 CTS 6.3.1 requires restart authorization when the reactor is shutdown due to exceeding a Safety Limit. CTS 6.3.2 requires the violation of a Safety Limit to be reported to the Commission. These requirements are not retained in Technical Specifications since they are a duplication of the regulations provided in 10 CFR 50.36(c)(1) and are not necessary to assure safe operation of the facility. The current regulations require ONS to perform all the actions currently required by Technical Specifications. As such, the proposed change is considered administrative and is consistent with the NUREG.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 The CTS applicability for the Reactor Core Safety Limits (CTS 2.1) is "during power operation." CTS 1.2.5 defines Power Operation as "...when the indicated neutron power range is above 2 percent of rated power as indicated on the power range channels." The ITS Applicability for Reactor Core Safety Limits is MODES 1 and 2. Thus, the ITS APPLICABILITY is more restrictive since MODE 2 is defined as ≤ 5% RTP with $k_{eff} \geq 0.99$. The inclusion of MODE 2 is appropriate because the reactor is critical in MODE 2 and limiting accidents and transients are postulated to begin in these MODES. The proposed change is consistent with the NUREG.

M2 CTS 6.3.1 requires the reactor to be shut down immediately and maintained in a safe shutdown condition until the Commission authorizes resumption of operation when a Safety Limit is violated. ITS 2.2.2 requires restoring compliance within limits when the Safety Limit for RCS Pressure (ITS 2.1.2) is exceeded. The addition of this requirement is appropriate in that it reduces the potential for exceeding the design pressure. The addition of this more restrictive action is consistent with the NUREG.

M3 CTS 6.3.1 does not establish specific required actions should the RCS Pressure Safety Limit be violated in MODES 3, 4, and 5. ITS 2.2.3 requires RCS pressure be restored within 15 minutes when the Safety Limit is violated. This represents a more restrictive requirement than that currently imposed. This more restrictive requirement is considered appropriate since exceeding the Safety Limit in MODE 3, 4, or 5 is potentially more severe than exceeding this Safety Limit in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. The proposed change is consistent with the NUREG.

TECHNICAL CHANGES - LESS RESTRICTIVE

L1 The CTS 2.2.1 applicability for the RCS Pressure Safety Limit (ITS 2.2.1) is "when there are fuel assemblies in the reactor vessel." The ITS applicability is MODES 1, 2, 3, 4 and 5. In essence, the ITS would be marginally less restrictive as it does not apply during MODE 6 while the CTS applies after the first assembly is 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. The proposed change is consistent with the NUREG.

L2 CTS 6.3 requires actions prescribed by 6.3.1 - 6.3.4 to be taken when a Safety Limit is violated. CTS 6.3.2 requires a Safety Limit violation to be reported to the Commission, the Site Vice President, and the Director of the Nuclear Safety Review Board. CTS 6.3.4 requires the report, with appropriate analyses and corrective action, to be submitted to the Commission within 10 days of the violation. The requirements of 6.3.2 and 6.3.4 are a duplication of reporting requirements described in 10 CFR 50.36(c)(1), 10 CFR 50.72, and 10 CFR 50.73 and are not included in the ITS. The CFRs are directly enforceable and removal of the reference to these regulations does not result in any decrease in requirements nor changes in methods of reporting. Therefore, removal of a reference to the CFR is considered administrative. However, 10 CFR 50.36(c)(1) requires the licensee to submit a Licensee Event Report (LER) as required by 10 CFR 50.73. The LER is not required until 30 days after occurrence of the event. Therefore, elimination of the 10 day report required by CTS and replacement with the 30 day report required by CFR is less restrictive. This is acceptable since the additional time allowed by CFR has no affect on the safety of the plant. The proposed change is consistent with the NUREG.

L3 CTS 6.3.1 requires the affected unit be shutdown immediately if a Safety Limit is exceeded. ITS Specifications 2.2.1 and 2.2.2 requires the affected Unit be placed in MODE 3 in 1 hour when a Safety Limit is exceeded. The ITS requirement is less restrictive since it requires the affected unit to be in MODE 3 within one hour where CTS requires the unit to be shut down immediately. This time period permits the shutdown to be performed in a more orderly and controlled manner than the current "immediately," while ensuring prompt remedial action is taken. This allows the Operator attention to be focused on restoring the Safety Limit rather than immediately placing the unit through a shutdown transient. The proposed change is consistent with the NUREG.

TECHNICAL CHANGES - REMOVAL OF DETAILS

LA1 The first and second paragraphs under the heading "Specification" in CTS 2.1 include information related to the method of assuring compliance with the Safety Limits for fuel pin centerline temperature and the departure from nucleate boiling ratio. The details of what constitutes compliance with a Safety Limit is relocated to the Bases for ITS 2.1.1. This detail is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for complying with the Safety Limit. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Therefore, relocation of this detail is acceptable. Changes to the Bases are controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications. The proposed change is consistent with the NUREG, as modified by TSTF-126.

LA2 CTS 6.3 requires actions prescribed by 6.3.1 - 6.3.4 to be taken when a Safety Limit is violated. CTS 6.3.2 requires a Safety Limit violation to be reported to the Commission, the Site Vice President, and the Director of the Nuclear Safety Review Board. The CTS 6.3.3 requirement to have the written report of the Safety Limit Violation reviewed by the Operations Superintendent and the Station Manager and submitted to the Site Vice President and the Director of the Nuclear Safety Review Board is relocated to the Quality Assurance Topical Report. The CTS 6.3.2 requirements for reporting the Safety Limit violation to the Site Vice President and the Director of the Nuclear Safety Review Board are relocated to the Quality Assurance Topical Report. These details are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these details to the Quality Assurance Topical Report provides reasonable assurance that the details are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Therefore, relocation of these details is acceptable. Changes to the Quality Assurance Topical Report are controlled by the provisions of 10 CFR 50.54. This change is consistent with the NUREG, as modified by TSTF-005, Revision 1.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 2.0 - SAFETY LIMITS

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS

ADMINISTRATIVE CHANGES

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." Some of the proposed changes involve reformatting, renumbering, and rewording of Technical Specifications. These changes, since they do not involve technical changes to the Technical Specifications, are administrative.

This type of change is connected with the movement of requirements within the current requirements, or with the modification of wording which does not affect the technical content of the current Technical Specifications. These changes will also include nontechnical modifications of requirements to conform to the Writer's Guide or provide consistency with the Improved Standard Technical Specifications in NUREG-1430. Administrative changes are not intended to add, delete, or relocate any technical requirements of the current Technical Specifications.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed changes involve reformatting, renumbering, and rewording of the existing Technical Specification. These modifications involve no technical changes to the existing Technical Specifications. The majority of changes were done in order to be consistent with NUREG-1430. During the development of NUREG-1430, certain wording preferences or English language conventions were adopted. The changes are administrative in nature and do not impact initiators of analyzed events. They also do not impact the assumed mitigation of accidents or transient events. Therefore, the changes do 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 changes involve reformatting, renumbering, and rewording of the existing Technical Specifications. The changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The changes will not impose any new or different requirements or eliminate any existing

requirements. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes involve reformatting, renumbering, and rewording of the existing Technical Specifications. The changes are administrative in nature and will not involve any technical changes. The changes will not reduce a margin of safety because it has no impact on any safety analysis assumptions. Also, since these changes are administrative in nature, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

MORE RESTRICTIVE CHANGES

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." Some of the proposed changes involve adding more restrictive requirements to the existing Technical Specifications by either making current requirements more stringent or by adding new requirements which currently do not exist.

These changes may include additional commitments that decrease allowed outage time, increase frequency of surveillance, impose additional surveillance, increase the scope of a specification to include additional plant equipment, increase the applicability of a specification, or provide additional actions. These changes are generally made to conform with the NUREG-1430.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event. If anything the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes. The changes do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, the changes do 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 changes provide more stringent requirements than previously existed in the Technical Specifications. The changes do not alter the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The changes do impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis.

Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. Adding more restrictive requirements either increases or has no impact on the margin of safety. The changes, by definition, provide additional restrictions to enhance plant safety. The changes maintain requirements within the safety analyses and licensing basis. As such, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGES - REMOVAL OF DETAILS

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." Some of the proposed changes involve moving details (engineering, procedural, etc.) out of the Technical Specifications and into a licensee controlled document. This information may be moved to the ITS Bases, UFSAR, plant procedures or other programs controlled by the licensee. The removal of this information is considered to be less restrictive because it is no longer controlled by the Technical Specification change process. Typically, the information moved is descriptive in nature and its removal conforms with NUREG-1430 for format and content.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed changes move details from the Technical Specifications to a licensee controlled document. The changes do not result in any hardware or operating procedure changes. The details being removed from the Technical Specifications are not assumed to be an initiator of any analyzed event. The licensee controlled documents containing the removed Technical Specification details are maintained using the provisions of 10 CFR 50.59, 10 CFR 50.54(a), 10 CFR 50.55(a), or other established review and control programs. Since changes to a licensee controlled document are evaluated per 10 CFR 50.59, 10 CFR 50.54(a), 10 CFR 50.55(a), or other established review and control programs, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated is involved. Therefore, the changes do 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 changes move detail from the Technical Specifications to a licensee controlled document. The changes will not alter the plant configuration (no new or different type of equipment will be installed) or make changes in methods governing normal plant operation. The changes will not impose different requirements, and adequate control of information will be maintained. The changes will not alter assumptions made in the safety analysis and licensing basis. Therefore, the changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes move detail from Technical Specifications to a licensee controlled document. The changes do not reduce the margin of safety since the location of details has no impact on any safety analysis assumptions. In addition, the details to be transposed from the Technical Specification to a licensee controlled document are the same as the existing Technical Specification. Future changes to this licensee controlled document will be evaluated per the requirements of 10 CFR 50.59, 10 CFR 50.54(a), 10 CFR 50.55(a), or other established review and control programs.

LESS RESTRICTIVE CHANGE L1

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> The CTS 2.2.1 applicability for the RCS Pressure Safety Limit (ITS 2.2.1) is "when there are fuel assemblies in the reactor vessel." The ITS applicability is MODES 1, 2, 3, 4 and 5. In essence, the ITS would be marginally less restrictive as it does not apply during MODE 6 while the CTS applies after the first assembly is 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. The proposed change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, the Oconee Nuclear Station has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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 has not been affected. 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 which is considered to be marginal. 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 an overpressure event are mitigated by the implementation of low temperature overpressurization protection requirements and administrative controls.

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.

LESS RESTRICTIVE CHANGE L2

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

CTS 6.3 requires actions prescribed by 6.3.1 - 6.3.4 to be taken when a Safety Limit is violated. CTS 6.3.2 requires a Safety Limit violation to be reported to the Commission, the Site Vice President, and the Director of the Nuclear Safety Review Board. CTS 6.3.4 requires the report, with appropriate analyses and corrective action, to be submitted to the Commission within 10 days of the violation. The requirements of 6.3.2 and 6.3.4 are a duplication of reporting requirements described in 10 CFR 50.36(c)(1), 10 CFR 50.72, and 10 CFR 50.73 and are not included in the ITS. The CFRs are directly enforceable and removal of the reference to these regulations does not result in any decrease in requirements nor changes in methods of reporting. Therefore, removal of a reference to the CFR is considered administrative. However, 10 CFR 50.36(c)(1) requires the licensee to submit a Licensee Event Report (LER) as required by 10 CFR 50.73. The LER is not required until 30 days after occurrence of the event. Therefore, elimination of the 10 day report required by CTS and replacement with the 30 day report required by CFR is less restrictive. This is acceptable since the additional time allowed by CFR has no affect on the safety of the plant. The proposed change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, the Oconee Nuclear Station has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change relaxes the time allowed to submit a report after a Safety Limit is violated from within 10 working days of the violation to within 30 days from discovery of the Safety Limit violation. This is consistent with the requirements in 10 CFR 50.73. This change will not result in operation that will increase the probability of initiating an analyzed event since the time frame for submitting an LER is not assumed in the initiation of any analyzed event. This change only affects the time frame for submitting the report after a Safety Limit is violated. This change will not alter assumptions relative to mitigation of an accident or transient event. This change will not alter the operation of process variables, structures, systems, or components as described in the safety analyses. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   This change relaxes the requirement for submitting a report to the NRC after a Safety Limit is violated. This change will not alter the plant configuration (no new or different type of equipment will be installed). This change only affects the time allowed to submit a report following a Safety Limit violation. This change does not impose different requirements; a report is still required. It will not alter assumptions made in the safety analysis. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change proposes to relax the time required for submittal of the report following a Safety Limit Violation. The time is extended from 10 working days of the violation to 30 days from discovery of the violation. Increasing the time for submitting a report does not affect the margin of safety since this change will not impact any safety analysis assumptions. As such, no question of safety is involved. Therefore, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L3

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 6.3.1 requires the affected unit be shutdown immediately if a Safety Limit is exceeded. ITS Specifications 2.2.1 and 2.2.2 requires the affected Unit be placed in MODE 3 in 1 hour when a Safety Limit is exceeded. The ITS requirement is less restrictive since it requires the affected unit to be in MODE 3 within one hour where CTS requires the unit to be shut down immediately. This time period permits the shutdown to be performed in a more orderly and controlled manner than the current "immediately," while ensuring prompt remedial action is taken. This allows the Operator attention to be focused on restoring the Safety Limit rather than immediately placing the unit through a shutdown transient. The proposed change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, the Oconee Nuclear Station has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   No significant increase in the probability or consequences of an accident is involved since the Completion Time for shutting down the reactor when a Safety Limit is violated is not assumed to be an accident precursor in a design basis accident. The extension of the Completion Time from immediately to 1 hour will have a negligible effect on the low probability an event occurring while the Safety Limit is not met and the plant is not shutdown. The proposed change allows 1 hour to shutdown the reactor in the event of a Safety Limit Violation. This time period permits the shutdown to be performed in a more orderly and controlled manner than the current "immediately," while ensuring prompt remedial action is taken. This allows Operator attention to be focused on restoring the Safety Limit rather than immediately placing the plant through a shutdown transient. Additionally, the consequences of an accident occurring during the proposed completion time are the same as the consequences of an accident occurring with the current shutdown requirements. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   This change will not alter the plant configuration (no new or different type of equipment will be installed). It will not alter assumptions made in the safety analysis. The proposed change only allows additional time to perform the shutdown. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Any reduction in a margin of safety will be insignificant since the change of the Completion Time does not affect any safety analysis assumption. Additionally, any reduction in a margin of safety will be offset by the benefit gained in allowing Operator attention to be focused on restoring the Safety Limit rather than immediately placing the plant through a shutdown transient. Therefore, this change does not involve a significant reduction in a margin of safety.

ENVIRONMENTAL ASSESSMENT

This proposed Technical Specification Change has been evaluated against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed changes meet the criteria for categorical exclusion as provided for under 10 CFR 51.22 (c) (9). The following is a discussion of how the proposed Technical Specification Change meets the criteria for categorical exclusion.

10 CFR 51.22 (c) (9): Although the proposed change involves changes to requirements with respect to inspection or surveillance requirements;

(i) the proposed change involves no Significant Hazards Consideration (refer to the No Significant Hazards Consideration section of this Technical Specification Change Request),

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite since the proposed changes do not affect the generation of any radioactive effluents nor do they affect any of the permitted release paths, and

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Based on the aforementioned and pursuant to 10 CFR 51.22 (b), no environmental assessment or environmental impact statement need be prepared in connection with issuance of an amendment to the Technical Specifications incorporating the proposed changes of this request.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 2.0 - SAFETY LIMITS

ATTACHMENT 5

NUREG 1430 MARKUP AND JUSTIFICATIONS

TECHNICAL SPECIFICATIONS

2.0 SAFETY LIMITS (SLs)

CTS

2.1 SLs

X Burnup 1

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

TSTF 126

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

TSTF 126

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. 3

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ ~~[~~2750~~]~~ psig. 2 2.2.1

2.2 SL Violations

With any SL violation, the following actions shall be completed: 6.3

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. 6.3.1

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. 3

(continued)

2.0 SLs

CTS

2.2 SL Violations (continued)

2.2.2 In MODE 1 or 2, if SL 2.1.2 is ~~not met~~ violated, restore compliance within limits and be in MODE 3 within 1 hour. (4) 6.3.1

2.2.3 In MODES 3, 4, and 5, if SL 2.1.2 is ~~not met~~ violated, restore RCS pressure to ≤ ~~[~~2750~~]~~ psig within 5 minutes. (2) DOC M4

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

~~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].~~ TSTF-005, R1

~~2.2.8 Operation of the plant shall not be resumed until authorized by the NRC.~~

3

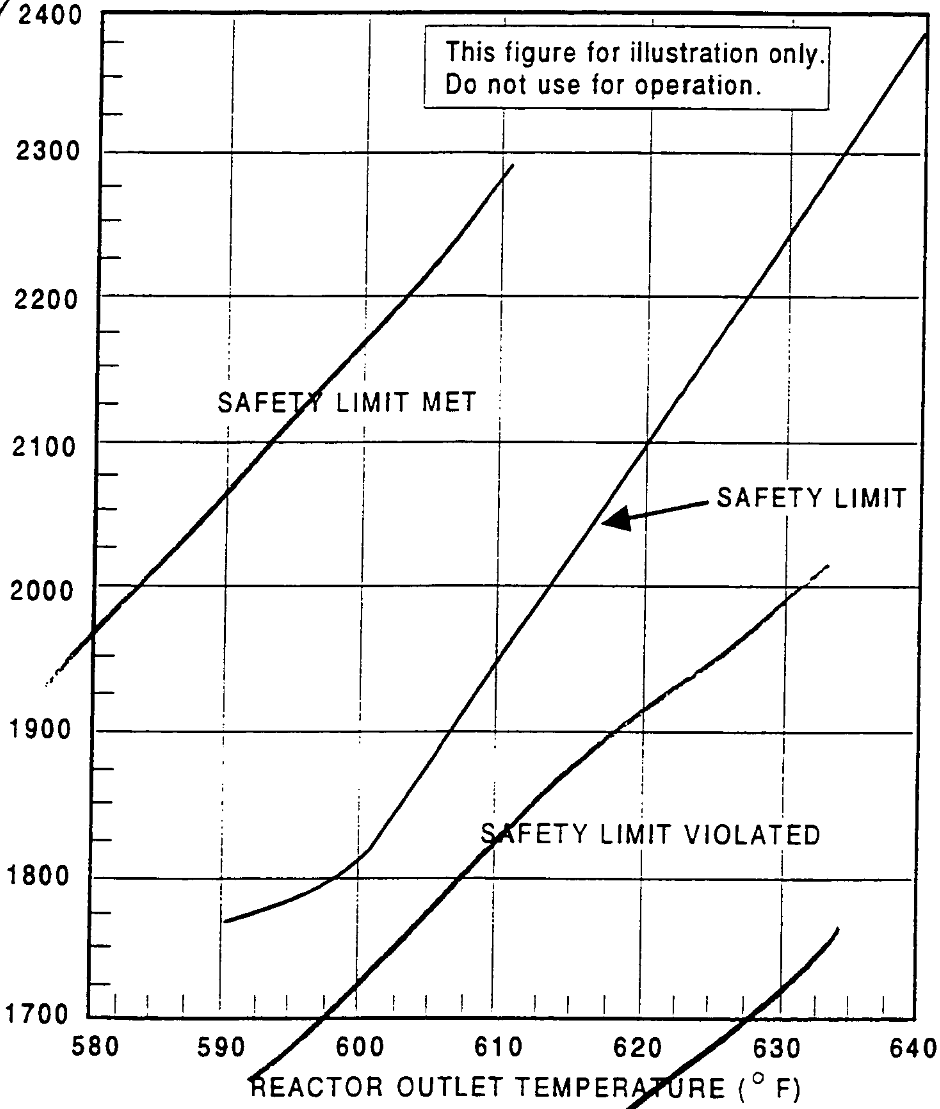

Figure 2.1.1-1 (page 1 of 1)
Reactor Coolant System Departure from Nucleate Boiling Safety Limits

TECHNICAL SPECIFICATIONS

1 The plant specific information from current technical specification (CTS) 2.1 for maximum local fuel pin centerline temperature was inserted in SL 2.1.1.1. This information is consistent with the ONS current licensing basis.

2 Brackets removed and appropriate plant specific information provided. CTS 2.1 currently specifies that the DNBR shall be maintained greater than the correlation limits of 1.3 for BAW-2 and 1.18 for BWC. However, since the BAW-2 correlation is no longer applicable (Oconee no longer uses Mark B fuel) it is not included in the ITS. Refer to Discussion of Change (DOC) A4 (Attachment 2 for this section).

3 NUREG Safety Limit 2.1.1.3 is actually a protective limit that ensures compliance with the Safety Limit (2.1.1.2) for departure from nucleate boiling ratio (DNBR). The second paragraph under the heading "Specification" in CTS 2.1 does refer to this protective limit as a method of complying with the Safety Limit for DNBR. However, the protective limit is not a Safety Limit. As such, NUREG 2.1.1.3 is not included in the ONS ITS. The pertinent information regarding its role in ensuring compliance with the DNBR Safety Limit is included in the ITS Bases for Safety Limit 2.1.1.2.

4 The wording in NUREG 2.2.3 and 2.2.4 was modified to be consistent with the wording used in NUREG 2.2.1. 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 Specifications. The proposed wording change is consistent with the other Standard Technical Specification NUREGs and Crystal River Unit 3 Technical Specifications, a B&W plant that has already converted to ITS.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 2.0 - SAFETY LIMITS

ATTACHMENT 6

NUREG 1430 MARKUP AND JUSTIFICATIONS

BASES

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

ONS Design Criteria 8

BASES

BACKGROUND

~~GDC 10~~ (Ref. 1) require~~s~~ 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)~~ transients 3. 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.

2 INSERT B2.0-1A

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. 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 to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

(continued)

INSERT B2.0-1A

DNB is not a directly measurable parameter during operation, but neutron power and Reactor Coolant System (RCS) temperature, flow and pressure can be related to DNB using a critical heat flux (CHF) correlation. The BWC (Ref. 2) CHF correlation has been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BWC correlation applies to Mark-BZ fuel. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual local 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 1.18 (BWC).

BASES

BACKGROUND (continued)

The proper functioning of the relief 13 Reactor Protection System (RPS) and main steam ~~safety~~ valves (MS~~S~~RVs) 13 prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and ~~AOOs~~ anticipated transients 3. 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 (Ref. ~~2~~ 3), in combination with all the LCOs, ~~is~~ are designed to prevent any ~~anticipated~~ analyzed combination of transient conditions for ~~Reactor Coolant System (RCS)~~ RCS temperature, flow and 11 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 DNB existence of flow instabilities. 1

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;

e. Reactor Coolant Pump to Power trip;

f. ~~Nuclear Overpower RCS Flow and Axial Power~~ Flux/Flow Imbalance trip; and 1

g. High Core Outlet Temperature trip; and 12

~~g.~~ h. MS~~S~~RVs. 13

The SL represents a design requirement for establishing the RPS trip setpoints identified previously.

(continued)

4 <Entire page>

BASES (continued)

SAFETY LIMITS

SL 2.1.1.1, and 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,~~ and ~~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 2.1.1.3 shows 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.~~

The SLs are preserved by monitoring ~~the~~ process variables, AXIAL POWER IMBALANCE, and Variable Low RCS Pressure to ensure that the core operates within the fuel design criteria. AXIAL POWER IMBALANCE and Variable Low RCS Pressure protective limits are provided in the COLR. The trip setpoints are derived by adjusting the measurement system independent AXIAL POWER IMBALANCE and Variable Low RCS Pressure protective limits given in the COLR to allow for measurement system observability and instrumentation errors.

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 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. 2

APPLICABILITY

SL 2.1.1.1 and 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.

R 13

(continued)

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.

4

2.2.1 ~~and 2.2.2~~

or

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. 4

unit 1

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.

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~~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).~~

~~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~~

(continued)

BASES

SAFETY LIMIT VIOLATIONS

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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.

REFERENCES

1. UFSAR, Section 3.1. ~~10 CFR 50, Appendix A, GDC 10.~~ 8

3 ~~2~~. USAR, Chapter 15 ~~Section [ ].~~ 1

3. 10 CFR 50.72.

4. 10 CFR 50.73. TSTF-005, R1

INSERT B 2.0-5A 2

INSERT B2.0-5A

2. BAW-10143P, Part 2, "Correlation of 15x15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation," August 1981.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

8 ONS Design Criteria

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~~ transients ~~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. 3

The design pressure of the RCS is 2500 psig. During normal operation and ~~AOOs~~ anticipated transients 3, 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). Hence, the safety limit is 2750 psig. To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure prior to initial operation, according to the ASME 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. 5 Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

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). The transient that is most influential for establishing the required relief capacity, and hence the valve size requirements and lift settings, is a rod withdrawal from low power. 7 During the transient, no control actions are assumed except that the safety valves on the secondary plant are assumed to open 6

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained. 6

The ~~overpressure protection analyses (Ref. 4) and~~ the safety analyses (Ref. 5) ~~are~~ performed using conservative assumptions relative to pressure control devices. is 7

7 limiting peak pressure transient, as determined by

More specifically, no credit is taken for operation of the following:

a. Pressurizer power operated relief valves (PORVs); 1

b. Steam line turbine bypass valves;

c. Control system runback of reactor and turbine power; and

d. Pressurizer spray valve.

(Ref. 2) 1

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 B31.7 (Ref. 6), is ~~120%~~ 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. 9

110%

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). 1

or steam generators

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 10

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the RCS pressure SL.

2.2.~~3~~ 2 — 4

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. ~~7~~ 6). 1

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.

2.2.~~4~~ 3 — 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.

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~~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).~~

(continued)

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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.~~

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28, 1988.~~ UFSAR, Section 3.1. 8

2. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000.

3. ASME Boiler and Pressure Vessel Code, Section XI, Article IW-5000.

~~4. BAW-10043, May 1972.~~

5. UFSAR, ~~Section [14]~~ Chapters 5 and 15.

~~6~~ 4. ASME USAS B31.~~1~~ 7, ~~Standard Code for Pressure~~ Nuclear Power Piping, dated February 19~~67~~ 68, with June 1968 Errata. 9

(continued)

BASES

REFERENCES (continued)

~~6~~ 7. 10 CFR 100. 1

~~8. 10 CFR 50.72.~~

~~9. 10 CFR 50.73.~~

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BASES

1 Editorial changes are made for clarity, preference or consistency with the Improved Technical Specifications (ITS) Writer's Guide. Renumbering and relettering are made as appropriate.

2 Specific detail relating to the critical heat flux correlation at ONS is included in the ITS B 2.1.1 Background information. This information is consistent with the ONS current licensing basis. Reference 2 has been added to reference the topical reports associated with the heat flux correlation.

3 Changes are made to reflect equivalent ONS terminology for anticipated operational occurrences (AOOs).

4 This change reflects changes made to the technical specifications.

5 The NUREG B 2.1.2 Background discussion description of the RCS inservice operational hydrotest at 100% design pressure is deleted. This type of testing is performed post modification and need not be discussed in the Bases of this specification.

6 The last sentence on page B2.0-6 of the NUREG is deleted as it does not accurately establish the plant conditions established in the ONS UFSAR Safety Analyses supporting the determination of required relief valve capacity. These plant conditions are established in the ONS UFSAR.

7 The next to last sentence on NUREG Bases page B 2.0-6 and the wording of the first full paragraph on page B 2.0-7 are revised to discuss the analyses in more general terms. In addition, the cited overpressure protection analyses were not the bases used and reference to them was deleted.

8 ONS was designed and licensed to the proposed Appendix A to 10 CFR 50, which was published in the Federal Register on July 11, 1967 (FR 32FR10213). Appendix A to 10 CFR 50 effective in 1971 and subsequently amended, is somewhat different from the proposed 1967 criteria. UFSAR section 3.1 includes an evaluation of ONS with respect to the proposed 1967 criteria. The NUREG statement concerning the GDC criteria is modified in the ITS to reference the current licensing basis description in the UFSAR.

9 The ONS 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 and renumbered.

10 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 of steam can be expected if the head is in place and not fully detensioned and removed. However, in agreement with the NUREG Bases, the amount of pressurization is not expected to be significant and thus the Specification should not be applicable in MODE 6.

11 The word "flow" is added since RCS flow is also a critical parameter in the DNB calculations.

12 The High Core Outlet Temperature trip is also a trip that automatically enforces the reactor core safety limits and is added to the list of trips described as fulfilling that function.

13 Changes are made to reflect equivalent ONS terminology for main steam safety valves (MSSVs). ONS uses main steam relief valves (MSRVs).

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 3.0 - LCO/SR APPLICABILITY

ATTACHMENT 1

TECHNICAL SPECIFICATIONS

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 12 hours;

b. MODE 4 within 18 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

(continued)

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.

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, 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 5.5.16, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.6 (continued)	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 LCO 3.1.8 allows 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

(continued)

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.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 3.0 - LCO/SR APPLICABILITY

ATTACHMENT 2

BASES

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

(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 Specification.

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

(continued)

BASES

LCO 3.0.2 (continued)

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.

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

(continued)

BASES

LCO 3.0.3 (continued)

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.

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.

(continued)

BASES

LCO 3.0.3 (continued)

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, "Spent Fuel Pool Water Level." LCO 3.7.14 has an Applicability of "During movement of irradiated fuel assemblies in the spent fuel 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 spent fuel pool" 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

(continued)

BASES

LCO 3.0.4 (continued)

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 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 allows 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 operating 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.

(continued)

BASES (continued)

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 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 a Required Action, 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

(continued)

BASES

LCO 3.0.6 (continued)

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 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 5.5.16, "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.

(continued)

BASES

LCO 3.0.6 (continued)

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.

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 LCO 3.1.8 allows 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) 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 an Exception LCO are only applicable when the Exception LCO is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

(continued)

BASES

SR 3.0.1 (continued)

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.

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 example of this process are:

a. Emergency feedwater (EFW) pump turbine maintenance during refueling that requires testing at steam pressures > 300 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 while 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

(continued)

BASES

SR 3.0.2 (continued)

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 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. 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 "inaccordance with 10 CFR 50, Appendix J, as modified byapproved exemptions." The requirements of regulations takeprecedence over the TS. The TS cannot in and of themselves extend a test interval specified in the regulations.

Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."

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.

(continued)

BASES

SR 3.0.2 (continued)

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 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

(continued)

BASES

SR 3.0.3 (continued)

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

(continued)

BASES

SR 3.0.4 (continued)

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 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.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 3.0 - LCO/SR APPLICABILITY

ATTACHMENT 3

CTS MARKUP AND DISCUSSION OF CHANGES

Specification 3.0

1.2.7 Refueling Operation

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

1.2.8 Startup

The reactor shall be considered in the startup mode when the shutdown margin is reduced with the intent of going critical.

1.3 OPERABLE

A system, subsystem, train, component or device shall be considered OPERABLE when it is capable of performing its intended safety functions. Implicit in this definition shall be the assumption that all essential auxiliary equipment required in order to assure performance of the safety function is capable of performing its related support function(s). Auxiliary equipment includes but is not limited to normal or emergency electrical power sources, cooling and seal water, instrumentation and controls, etc. If either the normal or emergency power to system, subsystem, train, component or device is not available it is considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (a) the alternate power source is available, ~~and (b) the redundant system is operable~~

<See 1.0>

<See 3.8>

A10

1.4 PROTECTIVE INSTRUMENTATION LOGIC

1.4.1 Instrument Channel

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 in nature.

1.4.2 Reactor Protective System

The reactor protective system is shown in Figures 7.2-1 and 7.2-4 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 protective channels, their associated instrument channel inputs, manual trip switch, all rod drive protective trip breakers and activating relays or coils.

<See 1.0>

1.4.3 Protective Channel

A protective channel as shown in Figure 7.2-1 of the FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply units, amplifiers and bistable modules provided for every reactor protective safety parameter) is a combination of instrument channels forming a single digital output to the protective system's coincidence logic. It includes a shutdown bypass circuit, a protective channel bypass circuit and reactor trip module and provision for insertion of a dummy bistable.

1 of 4

Specification 3.0

3 LIMITING CONDITIONS FOR OPERATION

3.0 LIMITING CONDITION FOR OPERATION

Specification

A2

LCO 3.0.3

In the event a Limiting Condition for Operation (LCO) and/or associated Action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the affected unit shall be placed in at least Hot Shutdown within the next 12 hours, and in at least Cold Shutdown within the following 24 hours unless corrective measures are completed that permit operation under the permissible Action statements for the specified time interval as measured from initial discovery or until the reactor is placed in a mode in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications.

M4

L1

A2

Bases

~~This specification delineates the ACTION to be taken for circumstances not directly provided for in the ACTION statements of existing LCOs and whose occurrence would violate the intent of the specification. For example, Specification 3.3.1 requires that two independent trains of the High Pressure Injection (HPI) System be operable and provides explicit Action requirements if one train of the HPI System is inoperable. Under the terms of Specification 3.0, if more than one train of the HPI System is inoperable, the affected unit is required to be in at least Hot Shutdown within the following 12 hours and in at least Cold Shutdown within the following 24 hours. It is assumed that the unit is brought to the required mode within the required times by promptly initiating and carrying out the appropriate Action statement.~~

A9

Add LCO 3.0.1 A3

Add LCO 3.0.2 A4

Add LCO 3.0.5 A5

Add LCO 3.0.6 A6

2 of 4

Add LCO 3.0.7 A7

A 89/89/86
12/10/80

M1

3.7 ELECTRICAL POWER SYSTEMS

LCO 3.0.4

~~TS 3.7.0~~

Condition

a MODE or other

LCO is

Entry into ~~operational~~ conditions ~~(e.g., HOT SHUTDOWN, COLD SHUTDOWN)~~ specified in the Applicability shall not be made when the ~~requirements of TS 3.7 are~~ not met, ~~unless~~ the associated ACTIONS ~~for the operational condition~~ to be entered permit continued operation in the ~~specified condition~~ for an unlimited period of time.

except when

MODE or other

This specification shall not prevent changes in ~~the operational~~ conditions specified in the Applicability which are required to comply with ACTIONS.

Other exceptions to this specification are stated in the individual specifications. These exceptions allow entry into ~~operational~~ conditions in the Applicability when the associated ACTIONS to be entered allow operation for only a limited period of time.

MODE or other Specified condition in the Applicability

MODES or other specified

or that are part of a shutdown of the unit

All

in the MODE or other specified condition in the Applicability

A1 <except as marked>

4 SURVEILLANCE REQUIREMENTS

4.0 SURVEILLANCE STANDARDS

~~**Applicability**~~

~~Applies to surveillance requirements which relate to tests, calibrations and inspections necessary to assure that the quality of structures, systems and components is maintained and that operation is within the safety limits and limiting conditions for operation.~~

~~**Objective**~~

~~To specify minimum acceptable surveillance requirements.~~

Specification

~~4.0.1~~ SR 3.0.1 Surveillance of structures, systems, components and parameters shall be as specified in the various subsections to this Technical Specification section, Section 4.0, except as permitted by Technical Specifications 4.0.2 and 4.0.3 below. M2

~~4.0.2~~ SR 3.0.2 Minimum surveillance frequencies, unless specified otherwise, may be adjusted as follows to facilitate test scheduling: M3

Specified Frequency	Maximum Allowable Interval Between Surveillances
Five times per week	2 days
Two times per week	5 days
Weekly	10 days
Bi-Weekly	20 days
Monthly	45 days
Bi-Monthly	90 days
Quarterly	135 days
Semiannually	270 days
Annually	18 months
18 months ~~Refueling Outage~~	~~22 months, 15 days~~ 22 1/2 months

~~4.0.3~~ SR 3.0.1 SR 3.0.4 If conditions exist such that surveillance of an item is not necessary to assure that operation is within the safety limits and limiting conditions for operation, surveillance need not be performed if such conditions continue for a length of time greater than the specified surveillance interval. Surveillance waived as a result of this specification shall be performed prior to returning to conditions for which the surveillance is necessary to assure that operation is within safety limits and limiting conditions for operation. A8

4.0.4 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)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components. <See 5.0>

<Add SR 3.0.3> M2

A 109/109/106
3/25/82

4 of 4

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with NUREG-1430. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1430. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1430. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A2 Current Technical Specification (CTS) Specification 3.0 is revised to adopt ITS Specification LCO 3.0.3 text:

a. The CTS phrase, "Exception to these requirements shall be stated in the individual specifications," is replaced with the phrase, "Exceptions to this Specification are stated in the individual Specifications," to clarify where exceptions to this LCO can be found.

b. The CTS phrase, "In the event a Limiting Condition for Operation (LCO) and/or associated Action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification," is replaced with the phrase, "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," to specifically state the circumstances which require compliance with this LCO.

c. The CTS phrase, "unless corrective measures are completed that permit operation under the permissible Action Statement for the specified interval as measured from initial discovery or until the reactor is placed in a mode in which the specification is not applicable," is replaced with the phrase "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," to clarify ambiguities regarding the termination of actions related to this LCO.

Since the ITS and CTS differ in wording and presentation only, these changes are considered to be administrative, and are consistent with the NUREG.

A3 A CTS Specification comparable to ITS LCO 3.0.1 does not exist. This Specification provides clarity with regard to when LCOs must be met, and where any exceptions can be found. Although not specifically stated in the CTS, this ITS Specification is consistent with CTS philosophy and application, and is therefore considered to be an administrative change. This change is consistent with the NUREG.

A4 A CTS Specification comparable to ITS LCO 3.0.2 does not exist. This Specification provides clarity with regard to the actions required to be taken upon discovery of a failure to meet an LCO. Although not specifically stated in the CTS, this ITS Specification is consistent with CTS philosophy and application, and is therefore considered to be an administrative change. This change is consistent with the NUREG.

A5 A CTS Specification comparable to ITS Specification LCO 3.0.5 does not exist. This Specification provides an exception to the NUREG Specification LCO 3.0.2 for those instances where restoration of equipment to an OPERABLE status could not be performed while continuing to comply with Required Actions. Many Technical Specification ACTIONS require that inoperable equipment be removed from service. To provide for performance of SURVEILLANCE REQUIREMENTS to demonstrate OPERABILITY of the equipment being returned to service, or to demonstrate OPERABILITY of other equipment, which otherwise cannot be performed without returning the equipment to service, the exception provided by the NUREG Specification LCO 3.0.5 is necessary. This Specification specifically establishes an allowance that is consistent with the intent of the CTS, and with accepted practice. Without this allowance, certain components could not be restored to OPERABLE status and a plant shutdown would ensue. It is not intended that Technical Specifications preclude the return to service of a component that is believed to be OPERABLE in order to confirm its OPERABILITY. This allowance is deemed to be a safer operation than requiring a plant shutdown to complete restoration and confirmatory testing. This change is therefore administrative, and is consistent with the NUREG.

A6 A CTS provision comparable to ITS LCO 3.0.6 does not exist. ITS LCO 3.0.6 provides guidance regarding the appropriate actions to be taken when a single inoperability (e.g., a support system) also results in the inoperability of one or more related systems (e.g., supported system(s)). In the CTS, along with their intent and interpretation provided by the NRC over the years, there is an ambiguous approach to the combined support/supported system inoperability. The NRC interpretations are described below.

- Guidance provided in the June 13, 1979, NRC memorandum from Brian K. Grimes (Assistant Director for Field Coordination) would indicate an intent/interpretation consistent with the proposed LCO 3.0.6 - without the necessity of also requiring the additional actions of a Safety Function Determination Program. That is, only the inoperable support system actions need be taken.

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

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

Considering the history of confusion and misunderstanding in this area, the BWOG STS, NUREG-1430, was developed with industry input and approval of the NRC to include LCO 3.0.6. The CTS provide guidance for losing normal or emergency power only. The new requirement encompasses each support systems, not just electrical power. Since previous guidance has been provided by the NRC and since the function of LCO 3.0.6 is to clarify existing ambiguities, and maintain actions within the realm of previous interpretations, this new provision is deemed to be administrative in nature.

A7 A CTS Specification comparable to ITS Specification LCO 3.0.7 does not exist. This Specification provides guidance with regard to Exceptions LCOs which allow certain Technical Specification requirements to be changed (i.e., made applicable in part or whole, or suspended) to permit performance of special tests or operations which otherwise could not be performed. This Specification eliminates confusion which would otherwise exist as to which LCOs apply during performance of a special test or operation. Although not specifically stated in the CTS, this ITS Specification is consistent with CTS philosophy and application, and is therefore considered to be an administrative change. This change is consistent with the NUREG.

A8 CTS 4.0.3 specifies that a surveillance need not be performed if conditions exist such that surveillance of an item is not necessary to assure that operation is within the safety limits and limiting conditions for operation. ITS SR 3.0.1 requires that SRs be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Additionally, ITS SR 3.0.1 specifies that surveillance need not be performed on inoperable equipment. CTS 4.0.3 requires surveillance to be performed prior to entering a condition for which the surveillance is necessary. ITS SR 3.0.4 requires applicable SRs be performed within their specified frequencies prior to entry into an LCO Applicability. This change is therefore administrative, and is consistent with the NUREG.

A9 The Bases of the current Technical Specifications for this section have been replaced by revised Bases that reflect the format and applicable content of the proposed Technical Specification Section 3.0, consistent with the NUREG-1430. The revised Bases are shown in the ITS.

A10 The portion of CTS 1.3 regarding requirements for the OPERABILITY of redundant systems when either normal or emergency power is not available to a system, subsystem, train component or device is not retained. CTS 1.3 provides that a system, subsystem, train, component or device may be considered OPERABLE for the purpose of satisfying the requirements of applicable LCOs provided either normal or emergency power is available and the redundant system is OPERABLE. Considering a system, subsystem, train component or device OPERABLE for the purpose of satisfying the requirements of applicable LCOs means that required actions specified in the applicable LCOs for the involved components are not required to be performed if the inoperability is due solely due to the loss of normal or emergency power. A similar provision is afforded for application of ITS LCO 3.0.6 to electric power and other support systems provided an evaluation is performed to determine whether the safety function can still be performed. By definition an OPERABLE redundant system is capable of fulfilling the safety function, albeit without consideration of an additional single failure. Therefore, this change is administrative in nature and is consistent with the NUREG.

A11 CTS 3.7 states that the specification does not prevent changes in the operational conditions specified in the Applicability which are required to comply with ACTIONS. ITS LCO 3.0.4 states 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. Although the CTS does not explicitly provide the exclusion associated with shutdown of a unit, its is not needed in the CTS presentation. CTS 3.7 is only applicable to the CTS electrical specifications (CTS 3.7.x). Each of the 3.7.x specifications is applicable in a MODE "Above Cold Shutdown." Since there are no interim operational conditions specified in the Applicability "Above Cold Shutdown," the only possible change in operating condition specified in the Applicability of the 3.7.x specifications results in exiting the Applicability for the Specifications. Therefore, this change is administrative in nature and is consistent with the NUREG.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.7 specifies limitations upon entry into an operational condition LCO when the requirements specified in the LCO for being in that operational condition are not met. CTS 3.7 is similar to ITS LCO 3.0.4 but is only applicable to CTS 3.7. ITS LCO 3.0.4 also establishes the requirement for the remainder of the ITS Specifications. ITS LCO 3.0.4 provides guidance related to MODE and operating condition entry when an LCO is not met. This Specification also clarifies those MODE changes permitted when required to comply with ACTIONS. For CTS requirements not contained in CTS 3.7, the CTS does not preclude entry into a MODE in which compliance with a Specification applicable to that MODE is not met at the time of entry. This change imposes more restrictive requirements and is consistent with the NUREG. The requirements of this change are reasonable and provide for a consistent and conservative approach to implementing the ITS requirements.

M2 CTS 4.0.1 requires that surveillances of structures, systems, components and parameters shall be as specified in the various subsections to CTS Section 4.0, except as permitted by Technical Specifications 4.0.2 and 4.0.3. ITS SR 3.0.1 requires SRs be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Additionally, 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. ITS SR 3.0.3 permits delaying applicable Actions resulting from the discovery an SR was not performed within its specified frequency. The requirements of this change are reasonable and provide for a consistent and conservative approach to implementing the ITS requirements.

CTS 4.0.1 does not explicitly require satisfactory completion of a surveillance within its specified frequency to meet the LCO. Since an explicit connection between the performance of a surveillance within the specified frequency and compliance with the LCO does not exist, the CTS does not impose an explicit time limit to complete the missed surveillance.

The explicit connection between satisfactory SR performance within the specified Surveillance Interval and Compliance with the LCO as well as the time limit to complete a missed SR are more restrictive requirements upon plant operation and are consistent with the NUREG.

M3 CTS 4.0.2 provides the maximum allowable interval between Surveillances. The specified intervals, with the exception of that for Refueling Outage

Frequency, is an approximate 50% extension to the base frequency. ITS SR 3.0.2 permits a 25% extension to the specified interval. The reduction in the extended interval from 50% to 25% is a more restrictive requirement upon plant operation and is consistent with the NUREG. The requirements of this change are reasonable and provide for a consistent and conservative approach to implementing the ITS requirements.

M4 In the event a Limiting Condition for Operation (LCO) and/or associated Action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, CTS 3.0 requires the affected unit shall be placed in at least Hot Shutdown within the next 12 hours, and in at least Cold Shutdown within the following 24 hours. ITS LCO 3.0.1 requires the unit be placed in MODE 3 within 12 hours, MODE 4 within 18 hours and MODE 5 within 37 hours. A CTS requirement comparable to the ITS LCO 3.0.3 requirement to be in MODE 4 within 18 hours does not exist and this change is consequently a more restrictive requirement upon unit operation. The requirements of this change are reasonable and provide for a consistent and conservative approach to implementing the ITS requirements.

TECHNICAL CHANGES - LESS RESTRICTIVE

L1 In the event a Limiting Condition for Operation (LCO) and/or associated Action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, CTS 3.0 requires the affected unit shall be placed in at least Hot Shutdown within the next 12 hours, and in at least Cold Shutdown within the following 24 hours. ITS LCO 3.0.1 requires the unit be in MODE 3 within 12 hours, MODE 4 within 18 hours and MODE 5 within 37 hours. Requiring the unit be placed in MODE 5 within 37 hours instead of 36 hours (12 hours plus 24 hours) is a less restrictive requirement upon unit operation and is consistent with the NUREG.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 3.0 - LCO/SR APPLICABILITY

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS

ADMINISTRATIVE CHANGES

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." Some of the proposed changes involve reformatting, renumbering, and rewording of Technical Specifications. These changes, since they do not involve technical changes to the Technical Specifications, are administrative.

This type of change is connected with the movement of requirements within the current requirements, or with the modification of wording which does not affect the technical content of the current Technical Specifications. These changes will also include nontechnical modifications of requirements to conform to the Writer's Guide or provide consistency with the Improved Standard Technical Specifications in NUREG-1430. Administrative changes are not intended to add, delete, or relocate any technical requirements of the current Technical Specifications.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed changes involve reformatting, renumbering, and rewording of the existing Technical Specification. These modifications involve no technical changes to the existing Technical Specifications. The majority of changes were done in order to be consistent with NUREG-1430. During the development of NUREG-1430, certain wording preferences or English language conventions were adopted. The changes are administrative in nature and do not impact initiators of analyzed events. They also do not impact the assumed mitigation of accidents or transient events. Therefore, the changes do 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 changes involve reformatting, renumbering, and rewording of the existing Technical Specifications. The changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The changes will not impose any new or different requirements or eliminate any existing

requirements. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed changes involve reformatting, renumbering, and rewording of the existing Technical Specifications. The changes are administrative in nature and will not involve any technical changes. The changes will not reduce a margin of safety because it has no impact on any safety analysis assumptions. Also, since these changes are administrative in nature, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

MORE RESTRICTIVE CHANGES

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." Some of the proposed changes involve adding more restrictive requirements to the existing Technical Specifications by either making current requirements more stringent or by adding new requirements which currently do not exist.

These changes may include additional commitments that decrease allowed outage time, increase frequency of surveillance, impose additional surveillance, increase the scope of a specification to include additional plant equipment, increase the applicability of a specification, or provide additional actions. These changes are generally made to conform with the NUREG-1430.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event. If anything the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes. The changes do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, the changes do 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 changes provide more stringent requirements than previously existed in the Technical Specifications. The changes do not alter the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The changes do impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis.

Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. Adding more restrictive requirements either increases or has no impact on the margin of safety. The changes, by definition, provide additional restrictions to enhance plant safety. The changes maintain requirements within the safety analyses and licensing basis. As such, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L1

The Oconee Nuclear Station is converting to the Improved Standard Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock & Wilcox Plants." The proposed change involves making the Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1431.

> In the event a Limiting Condition for Operation (LCO) and/or associated Action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, CTS 3.0 requires the affected unit shall be placed in at least Hot Shutdown within the next 12 hours, and in at least Cold Shutdown within the following 24 hours. ITS LCO 3.0.1 requires the unit be placed in MODE 3 within 12 hours, MODE 4 within 18 hours and MODE 5 within 37 hours. Requiring the unit be placed in MODE 5 within 37 hours instead of 36 hours (12 hours plus 24 hours) is a less restrictive requirement upon unit operation and is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The change will not allow continuous operation such that a single failure can preclude the associated function from being performed. The change permits one additional hour to place the unit in MODE 5. The probability of an accident occurring is not significantly affected by the small increase in the time to achieve MODE 5. Additionally, any increased risk resulting from the additional hour to achieve MODE 5 is partially offset by the reduced time allowed to place the unit in MODE 3. The consequence of an accident occurring are no greater during the additional hour permitted to achieve MODE 5 than in the 36 hours currently allowed.

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

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

The increase in time permitted to achieve MODE 5 represents a small increase in the time allowed to place the unit in an operating MODE where the initiating condition poses the least risk. The reduction in margin (time permitted with the unit exceeding the LCO or associated Required Action or Completion Time) is small and is partially offset by an increase in margin resulting from the reduced time allowed to place the unit in MODE 3. Therefore, the change does result in a significant reduction in the margin of safety.

ENVIRONMENTAL ASSESSMENT

This proposed Technical Specification Change has been evaluated against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed changes meet the criteria for categorical exclusion as provided for under 10 CFR 51.22 (c) (9). The following is a discussion of how the proposed Technical Specification Change meets the criteria for categorical exclusion.

10 CFR 51.22 (c) (9): Although the proposed change involves changes to requirements with respect to inspection or surveillance requirements;

(i) the proposed change involves no Significant Hazards Consideration (refer to the No Significant Hazards Consideration section of this Technical Specification Change Request),

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite since the proposed changes do not affect the generation of any radioactive effluents nor do they affect any of the permitted release paths, and

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Based on the aforementioned and pursuant to 10 CFR 51.22 (b), no environmental assessment or environmental impact statement need be prepared in connection with issuance of an amendment to the Technical Specifications incorporating the proposed changes of this request.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 3.0 - LCO/SR APPLICABILITY

ATTACHMENT 5

NUREG 1430 MARKUP AND JUSTIFICATIONS

TECHNICAL SPECIFICATIONS

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. TSTF-6, R1 Doc A3

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. Doc A4

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

a. MODE 3 within ~~7~~ 12 hours; 10

b. MODE 4 within ~~13~~ 18 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 3.7

(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. 4.0.3

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.~~ TSTF 104

LCO ~~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. 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.]~~ 9

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. DOC AS

(continued)

3.0 LCO APPLICABILITY (continued)

Doc A6

TSTF 166

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~~ shall be performed in accordance with Specification 5.5.~~15~~ 16, "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.

2

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.

5

Doc A7

LCO 3.0.7 Test Exception LCO~~s~~ ~~[~~3.1.9 (8) ~~, 3.1.10, 3.1.11 and 3.4.19]~~ allows 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. 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. TSTF 42

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.

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. DOC M2

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)

3.0 SR APPLICABILITY

SR 3.0.3 (continued) declared not met, and the applicable Condition(s) must be entered.

SR 3.0.4 — 4.0.3

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.~~] 9

TECHNICAL SPECIFICATIONS

NOTE: The first four justifications for these changes from NUREG-1430 were generically used throughout the individual LCO section markups. Not all generic justifications are used in each section.

1. The brackets are removed and the proper plant specific information or value is provided.

2. Editorial changes are made for clarity or for consistency with the Improved Technical Specifications (ITS) Writer's Guide.

3. The requirement/statement are deleted since it is not applicable to this facility. The following requirements are renumbered, where applicable, to reflect this deletion.

4. Changes are made (additions, deletions, and/or changes to the NUREG) to reflect the facility specific nomenclature, number, reference, system description, or analysis description.

5 NUREG LCO 3.0.7 refers to LCOs 3.1.10, 3.1.11 and 3.4.19. These LCOs are not in NUREG-1430. Therefore, reference to them has been deleted. [ONS-012]

6 Not used.

7 Not used.

8 Not used.

9 Bracketed reviewers Note is deleted.

10 LCO 3.0.3 requirements regarding the time to be in MODE 3 is changed to 12 hours to reflect the CLB. Should a Condition require multiple unit shutdown, the additional time provides for minimizing challenges to plant systems and personnel associated with a multi unit shutdown. Additionally, for a single unit shutdown, the 12 hours provides some additional time to potentially correct any Condition necessitating the shutdown. Any increased risk associated with the extension of time to be in MODE 3 is at least partially offset by a reduction in risk associated with averted plant shutdowns and the averted potential for shutdown transients. Additionally, the LCO 3.0.3 time to attain MODE 4 is modified to 18 hours to reflect the additional time allowed to attain MODE 3. The time to attain MODE 5 is not modified.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 3.0 - LCO/SR APPLICABILITY

ATTACHMENT 6

NUREG 1430 MARKUP AND JUSTIFICATIONS

BASES

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 through LCO 3.0.~~8~~ 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

(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 Specification~~s.~~ 2

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."

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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.

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. 2

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." LCO 3.7.14 has an Applicability of "During movement of irradiated fuel

(continued)

BASES the 2

LCO 3.0.3 (continued)

assemblies in 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 fuel storage pool" 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.

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)

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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.

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Exceptions to LCO 3.0.4 are stated in the individual Specifications. ~~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 ~~only while~~ operating 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. ~~[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.]~~ (2) (8)

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 B 3.0-6A

The exceptions allows 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:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

required testing to demonstrate OPERABILTY

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required testing

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.

2

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 Action(s) and must be reopened to perform the ~~SRs~~.

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. 2

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~~ 16, "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.
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 LCO~~s~~ ~~[~~3.1.~~9~~ 8 ~~[illegible]~~ allows 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. 1
	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) 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 an ~~Special Test~~ Exception ~~(STE)~~ LCO are only applicable when the ~~STE~~ LCO is used as an allowable exception to the requirements of a Specification. 7

Exception

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.

INSERT B3.0-10A

TSTF-8, R2

(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)

TSTF 43

Insert B3p-11A

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.

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 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. 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.

(continued)

INSERT B 3.0-11A

Some example of this process are:

a. ~~Auxiliary~~ Emergency feedwater (~~A~~EFW) pump turbine maintenance during refueling that requires testing at steam pressures > 300 psi. However, if other appropriate testing is satisfactorily completed, the ~~A~~EFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed while the plant reaches the steam pressure required to perform the ~~A~~EFW pump testing. E-4

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."

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.

Satisfactory ~~C~~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. 2

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 ~~LCO~~ SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. 2

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.

associated with operation

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. 2

Bases

NOTE: The first five justifications for these changes from NUREG-1430 were generically used throughout the individual Bases section markups. Not all generic justifications are used in each section.

1 The brackets are removed and the proper plant specific information or value is provided.

2 Editorial changes are made for preference, clarity or consistency with the Improved Technical Specifications (ITS) Writer's Guide.

3 The requirement/statement are deleted since it is not applicable to this facility. The following requirements are renumbered, where applicable, to reflect this deletion.

4 Changes are made (additions, deletions, and/or changes to the NUREG) to reflect the facility specific nomenclature, number, reference, system description, or analysis description.

5 This change reflects changes made to the technical specifications.

6 Not used.

7 The bases is revised for consistency with the scope and content of the associated specification.

8 The bracketed portion of the Bases for LCO 3.0.4 is not adopted since there is no instance of the associated Note in the ITS.

9 The portion of TSTF-52 which modifies the Bases example where SR 3.0.2 does not apply is not adopted. ONS has adopted 10 CFR 50, Appendix J, Option B for Type A leakage rate testing only. The requirements of 10 CFR 50, Appendix J, Option A remain applicable for Type B and C leakage rate testing.