General Electric Advanced Technology Manual

Chapter 3.0

Technical Specifications

TABLE OF CONTENTS

LIST OF TABLES

ATTACHMENTS

3.0 TECHNICAL SPECIFICATION ORGANIZATION

Learning Objectives:

1. Recognize the purpose of the station's Technical Specifications.

2. Recognize the attributes of the three different Technical Specification formats currently in use.

3. Recognize the meaning of the terms listed in the Improved Standard Technical Specifications (ISTS) in Section 1.1, Definitions.

4. Given the ISTS and a specific set of plant conditions, determine which MODE the unit is operating in.

5. Given the Improved Standard Technical Specifications and the TS Bases, a specific set of plant conditions, and system training, apply the following ISTS sections to determine the required actions of the licensee:
   a. Definitions in section 1.1
   b. The use of Logical Connectors in section 1.2
   c. Completion Time guidance in section 1.3
   d. Surveillance Frequency guidance in section 1.4
   e. Safety Limit guidance of section 2.0
   f. General Limiting Condition of Operations (LCO) guidance of section 3.0
   g. General Surveillance Requirements (SR) guidance of section SR 3.0
   h. Specific LCO and SR guidance of sections 3.1 through 3.10
   i. Design Feature guidance of section 4.0
   j. Administrative Controls guidance of section 5.0

3.0.1 Introduction

The purpose of Technical Specification is to protect the health and safety of the public by imposing limits, operating conditions, and other similar requirements on the facility. The legal requirements for plant technical specifications are found in 10 CFR 50.36 which states "The technical specifications will be derived from the analyses and evaluation included in the safety analysis report...." Paraphrasing this statement, technical specifications define the limits of plant operation to ensure that the plant is operated within those boundaries established by the Safety Analysis. For example, if the safety analysis uses a maximum reactor coolant system pressure of 1325 psig then a technical specification limit of 1325 psig will be imposed. After the plant=s technical specifications have been approved by the Commission, they become an attachment to the Facility Operating License.

3.0.2 Derivation

The format for technical specifications evolves from 10 CFR 50.36 which lists the following sections to be included in technical specifications:

- Safety limits and Limiting safety system settings,
- Limiting conditions for operation,
- Surveillance requirements,
- Facility Design features, and
- Administrative controls.

For special items of interest, the NRC issues Regulatory Guides which describe methods acceptable to the NRC staff of implementing specific parts of regulations. Regulatory Guide 1.70, issued in 1972, provided the STANDARD format and content of Safety Analysis Reports (SARs). This guide specified seventeen chapters in the SAR and it assigns technical specifications to Chapter 16. The Standard Review Plan, NUREG 0800, contains the current guidance for review of SARs, including advanced reactors. Portions of these Regulatory Guides dealing with technical specifications are included below.

3.0.3 Format

Three different technical specification formats are currently in use at licensed facilities: custom, standard, and improved standard technical specifications. These formats are discussed in the following paragraphs.

3.0.3.1 Custom Technical Specifications

Originally, technical specifications were prepared on an individual basis for each facility and thus became known as custom technical specifications. This ad hoc approach resulted in the issuance of specifications which addressed each of the categories required by 10 CFR 50.36, but also resulted in great diversity in terms of the technical content of specifications and the interpretations of requirements by licensee staffs and NRC inspectors. Although custom technical specifications have largely been supplanted by standard formats throughout the commercial nuclear power industry, they remain in use at a few plants. Table 3.0-1 provides an example of the "custom" format for the Automatic Depressurization System Limiting Condition for Operation (LCO) at Nine Mile Point.

3.0.3.2 Standard Technical Specifications

In an effort to provide a systematic approach to technical specification content, the NRC initiated the Standard Technical Specification Program in the 1970s. This program resulted in the issuance of standard technical specifications for each nuclear steam supply system (NSSS) design. For General Electric plants, the standard format and content of technical specifications was provided in NUREG-0123, “Standard Technical Specifications for General Electric Boiling Water Reactors”. The standard technical specifications were used as the template for the technical specifications of newly licensed plants, and many licensees converted their custom technical specifications to the standard format as well. Table 3.0-2 provides an example of the standard technical specification format for Shutdown Margin Limiting Condition for Operation (LCO) at Limerick.

3.0.3.3 Improved Standard Technical Specifications

The third type of technical specifications is the Improved Standard Technical Specifications (ISTS) contained in NUREG-1433. Revision 3 of NUREG-1433 is the subject of this manual chapter and is the version of the STS that will be used in both the R504B and R624B courses. The improved STS were originally issued in April of 1995 and have been subsequently revised to incorporate the changes resulting from the experience gained from license amendment applications. Many licensees have or plan to convert to these improved Standard Technical Specifications or to adopt partial improvements to existing technical specifications. NUREG-1433 was the result of extensive public technical meetings and discussions between the Nuclear Regulatory Commission staff and various nuclear power plant licensees, Nuclear Steam Supply System (NSSS) Owners Groups, specifically the GE Owners Group, NSSS vendors, and the Nuclear Energy Institute. Licensees are encouraged to upgrade their technical specifications consistent with those criteria and conforming, to the extent practical and consistent with the licensing basis for the facility, to the improved STS. Licensees adopting portions of the improved STS to existing technical specifications should adopt all related requirements, as applicable, to achieve a high degree of standardization and consistency.

One of the advantages to the licensees in adopting the improved STS is that the NRC can approve generic changes to the specifications and the utility can reference those changes in their specific license amendment request. An industry group known as the Technical Specification Task Force (TSTF) will generate a generic change which then goes through the review and approval process. Each proposed change is given a unique number and a TSTF follower is generated to track the request through the approval process. The licensee can then reference that TSTF number which greatly reduces the approval time for the amendment.

The improved STS consist of two volumes: Standard Technical Specifications and the Standard Technical Specifications Bases.

The Technical Specifications volume, illustrated in Table 3.0-3, begins with the LCO followed by the applicability, action, and surveillance sections. The actions sections are divided into three columns (condition, required action, and completion time) while the surveillance sections are divided into two sections (surveillance and frequency). This format is provided to better articulate to the operator the conditions that exist and what must be performed for that condition. Since these STS are generic in nature, the actual values listed in the specifications or even the equipment covered, may vary depending on the specific plant configuration. These values and/or equipment differences are listed in brackets.

The Bases volume includes descriptive and background information that defines and amplifies the reasoning and importance of the specification and surveillance requirement. It may further better define exactly what may constitute operability. It is imperative that the bases information ALWAYS be reviewed before reaching a final conclusion on any related issue. The TS Bases can be revised without a license amendment, as long as the change does not constitute an unreviewed safety question as outlined in 10 CFR 50.59.

3.0.3.4 Technical Requirements Manual

Since technical specification revisions are costly and can take considerable time to be approved, it is advantageous to remove items that may not meet the specific criteria of 10 CFR 30.36 and/or can be effectively controlled under another administrative process. One example is the fire suppression systems, which were originally required under the STS, but now are controlled under other administrative processes which are referenced in the operating license. Another example would be the plant heat up and cooldown restrictions which may have been removed from the Technical Specifications and are now controlled under the Plant Temperature Limit Report (PTLR).

Many utilities maintain a Technical Requirements Manual (TRM) which contains many of the systems formally listed in the TSs. The TRM normally has specific specifications and their bases in the same format as the improved STS. They may also contain operational conveniences, such as lists, cross references, acceptance criteria, and drawings. This volume was developed as a matter of convenience to reduce the number of inconsequential Technical Specification revisions required by the other formats. The TRM may also contain the Core Operating Limits Report (COLR) and/or PTLR as an attachment. The TRM provides the licensee flexibility in modifying the TRM functions via controlled processes without requiring a NRC approved Tech Spec amendment. The Technical Requirements Manual is laid out and used very similar to the direct use of Technical Specifications.

3.0.3.5 Risk Informed Technical Specifications

Since the mid-1980s, the NRC has been reviewing and granting improvements to technical specifications that are based, at least in part, on probabilistic risk assessment (PRA). The Commission reiterated that it expects licensees to use any plant-specific PRA or risk survey in preparing technical specifications for NRC approval. In August 1995, the NRC adopted a final policy statement on the use of PRA methods in nuclear regulatory activities that encourages greater use of PRA to improve safety decision making and regulatory efficiency. These improvements are intended to maintain or improve safety while reducing unnecessary burden and to bring technical specification requirements into congruence with the Commission's other risk-informed regulatory requirements, in particular, the maintenance rule. Several of these risk informed initiatives have already been adopted. Examples include:

- Allowing entry into a mode or condition when not all required equipment is operable as long as a risk assessment is completed and risk management actions are established (LCO 3.0.4).
- Allowing delaying completion of a missed surveillance without declaring the system inoperable as long as a risk evaluation is performed and the risk impact is managed (SR 3.0.3).
- Allowing inoperability of one or more snubbers for a specified period of time without declaring the supported system inoperable if the risk is assessed and managed (LCO 3.0.8).

Revision 4 of the Standard Technical Specifications (NUREG-1433) has been issued and several generic technical specification revisions have been approved which incorporate Risk Informed Technical Specifications (RITS) initiatives. Although these approaches have been approved generically, very few licensees have received site specific approval for the changes. These changes are presented here to familiarize the student with these changes although this course uses revision 3 of NUREG-1433. The major changes are discussed below.

Initiative 1 modified the end-states for many LCOs to allow plants to remain in Mode 3 and not cooldown. Table 3.0-4 provides an example of the changes approved under this initiative. Revision 4 of NUREG-1432 does contain these modified end states.

Initiative 4b allows the utility to establish risk-informed completion time based on the unit specific probabilistic risk assessment (PRA) under a Risk Informed Completion Time (RICT) Program. This program is included in the TS by reference in Section 5 of the TS. The program must be implemented in accordance with NEI-06-09, “Risk Managed Technical Specification (RMTS) Guidelines”. When adopted, the completion time will include both a specific time (e.g., 8 days) or the utility can use a risk-informed completion time that is greater than the listed completion time if justified by the change in risk determined by the PRA. Several restrictions are applied to RICTs:

- The RICT may not exceed 30 days.
- A RICT may only be utilized in MODE 1, 2 , and MODE 3 while relying on the main condenser for heat removal;
- When a RICT is being used, any plant configuration change within the scope of the Risk Informed Completion Time Program must be considered for the effect on the RICT.
  - For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.
  - For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.
- Use of a RICT is not permitted for voluntary entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE.
- Use of a RICT is permitted for emergent conditions which represent a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE if one or more of the trains are considered "PRA functional" as defined in Section 2.3.1 of NEI 06-09.

Table 3.0-5 provides an example of the changes approved under this initiative. Revision 4 of NUREG-1432 does not include the use of RICTs, but TSTF-505 has been approved by the NRC and is available for use. This initiative has been piloted in the industry, but to date no utility has adopted the change.

Initiative 5b allows utilities to establish surveillance requirement frequency under a program outside of technical specifications. This approach has already been used in a number of programs (e.g., Inservice Testing Program, Primary Containment Leakage Rate Testing Program and Ventilation Filter Testing Program). Table 3.0-6 provides an example of the changes approved under this initiative. Revision 4 of NUREG 1432 does contain the option to relocate SR frequency determination outside of TSs. Duane Arnold has already adopted this change.

Initiative 7 provides guidance on the impact of non-tech spec support systems on the OPERABILITY of TS supported systems. As stated earlier, the impact of inoperable snubbers on the supported systems has previously been approved and incorporated into revision 3 of NUREG-1433. The impact of the inoperable (or missing) barriers on the OPERABILITY of their supported TS systems is also addressed by this initiative. Barriers are doors, walls, floor plugs, curbs, hatches, installed structures or components, or other devices, not explicitly described in Technical Specifications that support the performance of the safety function of systems described in the Technical Specifications. This change is included in revision 4 of NUREG-1433. It allows a delay in declaring the supported system inoperable for up to 30 days if the risk is assessed and managed. An example of this change has been adopted at River Bend and is illustrated in Table 3.0-7.

3.0.4 Improved Standard Technical Specifications

The description that follows is based on the new revised standard technical specification format (the third format discussed above) and will be used in the Advanced Technology and Simulator Courses. Where bracketed [ ] information is presented, it indicates that the information is plant specific and therefore variable up to and including not applicable to a specific unit.

3.0.4.1 Use and Application (Section 1.0)

This section of the technical specifications manual is comprised of four subsections:

- Definitions (Section 1.1)
- Logical Connectors (Section 1.2)
- Completion Times (Section 1.3)
- Frequency (Section 1.4)

Subsection 1.1 provides defined terms that appear in capitalized type and are applicable throughout technical specifications and bases. Wherever these terms appear throughout STS they will also be in capitalized type to cue the user of the specific definition that is in section 1.1. Some key terms of frequent use include:

- CORE ALTERATION
- CORE OPERATING LIMITS REPORT (COLR)
- LEAKAGE
- MODE
- OPERABLE-OPERABILITY
- RATED THERMAL POWER
- SHUTDOWN MARGIN (SDM)
- THERMAL POWER

Using a copy of STS (NUREG-1433), review these definitions.

The Logical Connectors, subsection 1.2, explains the meaning of logical connectors and provides examples to illustrate their usage. The logical connectors of use are the words <u>AND</u> and <u>OR</u> and will appear in the Required Action and Surveillance portion of the specification. The connectors are always in all caps and underlined. The key to their use is the level of indentation in which they appear in the specification. Likewise, the numbering of the action or surveillance can be used to reinforce the intent of the connector. Using a copy of STS (NUREG-1433), review the logical connector examples in section 1.2-1 and 1.2-2.

Subsection 1.3, Completion Times, establishes the completion time convention and provides guidance for its use. The completion times are the time allowances for completing a required action to maintain compliance with the technical specification requirements. The information in Section 1.3 enhances and imposes specific conditions on the completion time requirements of specific required actions. Using a copy of STS (NUREG-1433), review the Section 1.3 completion time guidance and examples related to:

- Exiting the applicability of the specification or repairing the equipment
- Separate condition entry allowed (ex. TS 3.1.3)
- Completion time extension due to subsequent subsystem inoperability (ex. TS 3.5.1 for subsequent inoperability of another RHR pump in that loop).
- Limitations on completion time extensions (ex. TS 3.1.3 and 3.8.1 Condition A and C)

Subsection 1.4, Frequency, defines the proper use and application of frequency requirements. The frequencies are the time intervals allowed between performances of the surveillances that demonstrate operability of the required equipment. Using a copy of STS (NUREG-1433), review the Section 1.4, frequency guidance and examples. Pay particular attention to the difference between surveillances being “met”, versus being “performed”.

3.0.4.2 Safety Limits and Limiting Safety System Settings (Section 2.0/3.0)

10CFR-50.36 states “Technical specifications will include items in the following categories:”

***Safety limits, limiting safety system settings, and limiting control settings*.**

Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity. Per 10 CFR 50.36:

- If any safety limit is exceeded, the reactor must be shut down.
- The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence.
- Operation must not be resumed <u>until authorized by the Commission</u>.
- The licensee shall retain the record of the results of each review until the

Commission terminates the license for the reactor.

This section of the plant technical specifications establishes the requirements for the protection of the fission product barriers.

For BWRs, the safety limits found in Section 2.0 are:

- Thermal Power, at Low Pressure or Low Flow
- Minimum Critical Power Ratio (MCPR), at High Pressure and High Flow
- Reactor Coolant Pressure
- Reactor Vessel Water Level

Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. (RPS, NSSSS, ECCS)

- Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.
- If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor.
- The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the reactor.

The BWR Limiting Safety System Settings are a subset of the Limiting Conditions for Operation specifications (Section 3.0-3.10) where ECCS, RPS and NSSS setpoints are found.

3.0.4.3 LCOs and Surveillance Requirements (Section 3.0)

10CFR-50.36 states “Technical specifications will include items in the following categories:”

Limiting Conditions for Operation

Limiting conditions for operation are the lowest functional capabilities or performance levels of equipment required for safe operation of the facility (i.e., able to perform the safety function).

- When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

- The licensee shall retain the record of the results of each review until the Commission terminates the license for the nuclear reactor or the fuel reprocessing plant. In the case of . . .

The limiting conditions for operation (LCO's) (Section 3.0-3.10) are requirements that must be satisfied for the unrestricted operation of the unit. The statements that follow the limiting condition for operation (LCO) are actions that must be taken in the event that the LCO cannot be satisfied.

Section 3.0 is used to establish the ground rules for the remaining portions of technical specifications LCO's (Sections 3.1-3.10). Using a copy of STS (NUREG-1433), review the following Section 3.0 LCO Applicability statements:

LCO-3.0.1 specifies requirements and exceptions to when LCO's must be met. Important to this is the "Applicability".

LCO-3.0.2 provides direction on what to do if an LCO is not met.

LCO-3.0.3 is one of the most important specifications. This specification provides guidance for plant operation when the LCO <u>and</u> its associated action statements cannot be satisfied or if you find yourself in a position where something must be done but no guidance exists. In summary, when the plant is less conservative than the least conservative technical specification action statement, go to specification 3.0.3.

LCO-3.0.4 provides restrictions on plant operational MODE changes when required equipment is not operable.

LCO-3.0.5 provides guidance on performing actions inconsistent with the required actions when needed to perform testing that demonstrates OPERABILITY.

LCO-3.0.6 provides exceptions from declaring supported systems inoperably solely because the support system is declared inoperable. This is commonly known as the "daisy chain" or "cascading" LCO.

LCO-3.0.7 provides requirements for and allows the use of the Special Operations LCO's of section 3.10

LCO-3.0.8 provides exception guidance on declaring supported systems inoperable solely based upon the inoperability of piping snubbers.

Sections 3.1-3.10 contain specifications specific to individual systems and will be discussed later in this chapter.

3.0.4.4 Surveillance Requirements (Section 3.0)

10CFR-50.36 states "Technical specifications will include items in the following categories:"

Surveillance requirements

Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met (i.e., demonstrate the ability to perform the safety function).

The Surveillance requirements and frequencies of performance are located with the LCO sections 3.0-3.10. Section 3.0 contains a subsection supporting specific ground rules related to surveillance testing and frequencies. Using a copy of STS (NUREG-1433), review the following Section 3.0 Surveillance Requirement statements:

SR-3.0.1 provides guidance on when surveillances must be "met" and actions if not met or not performed.

SR-3.0.2 provides a grace period for exceeding the required surveillance frequency due to unforeseen circumstances. It is not meant for operational convenience.

SR-3.0.3 provides relief on declaring equipment inoperable if it is found that a surveillance requirement was not conducted within its required frequency. It is not meant for operational convenience nor is it intended for a surveillance that has been found to never have been performed.

SR-3.0.4 compliments LCO-3.0.4 by restricting MODE changes unless the applicable surveillance testing has been completed for the new MODE.

Sections 3.1-3.10 contain surveillance requirements and frequencies specific to individual systems and will be discussed later in this chapter.

3.0.4.5 Design Features (Section 4.0)

10CFR-50.36 states "Technical specifications will include items in the following categories:"

Design Features

Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in the categories of Safety Limits and

Limiting Safety System Settings, Limiting Conditions for Operation and Surveillance Requirements.

Section 4.0 describes the important design features of the unit. Items such as the allowable numbers of control rods and fuel bundles in the core as well as spent fuel pool requirements are to be found here. Using a copy of STS (NUREG-1433), review the Section 4.0 Design Features.

3.0.4.6 Administrative Controls (Section 5.0)

10CFR-50.36 states “Technical specifications will include items in the following categories:”

Administrative Controls

Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Subsections include:

- Responsibility, Organization (including Shift Staffing) and Staff Qualifications
- Procedures
- Programs and Manuals
- Reporting Requirements
- High Radiation Areas

Administrative controls delineate the management and staff organization, qualification, review and audit groups, record and reporting requirements, and procedures required to assure safe plant operation. The administrative organization is addressed in terms of offsite management and onsite staff requirements including the minimum shift crew composition for all plant conditions.

Programmatic controls are technical specification requirements on licensee processes such as:

- Offsite Dose Calculation Manual
- Post-Accident Sampling
- Radioactive Effluent Controls
- Inservice Testing Program
- Ventilation Filter Testing Program
- TS Bases Control Program
- Safety Function Determination Program
- Primary Containment Leakage Rate Program

Using a copy of STS (NUREG-1433), review Section 5.0 Administrative Controls.

3.0.5 LCO (Sections 3.1-3.10)

A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

Criterion 1

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

- Drywell Equipment and Floor Drain Sump Level Monitoring

Criterion 2

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

- Drywell Pressure/Suppression Pool Level
- APRM Scram

Criterion 3

A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

- Emergency Core Cooling Systems
- Reactor Protection System Instrumentation

Criterion 4

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

- ATWS Instrumentation (RPT and ARI)

The following is a listing of the sections or categories and their associated systems:

3.1 Reactivity Control Systems
3.2 Power Distribution Limits
3.3 Instrumentation
3.4 Reactor Coolant System
3.5 Emergency Core Cooling System and Reactor Core Isolation Cooling System
3.6 Containment Systems
3.7 Plant Systems
3.8 Electrical Power Systems

3.9 Refueling Operations
3.10 Special Operations

The information that follows contains more detail related to each of the LCO Sections 3.1 through 3.10. Attachment 1 contains some guidance on how to approach and resolve any tech spec related problem. Your success at arriving at the correct answer is significantly related to your understanding of the equipment and knowledge of its purpose(s) and operation. Attachment 2 contains some examples of application of the requirements and surveillances to the individual topical areas. These examples will be covered during the individual TS section lectures. You are encouraged to look ahead and attempt resolution of these examples. The answers to these example problems will be distributed following each lecture.

3.0.5.1 Reactivity Control Systems (Section 3.1)

This section addresses the equipment and conditions for the reactivity control systems required to safely control and shutdown the reactor. The section includes:

3.1.1; SHUTDOWN MARGIN:

As defined in section 1.1 and applied in 3.1.1, there is a minimum excess amount of negative reactivity that the licensee must be capable of inserting to ensure the reactor is and will remain shutdown.

3.1.2; Reactivity Anomalies:

This requirement basically states that the reactivity effect of the control rods going in or out of the core must be predictable to some level of accuracy.

3.1.3; Control Rod OPERABILITY:

With the application of the section 1.1 definition of OPERABILITY this LCO addresses the loss of safety function of the control rods for various reasons and considers the requirements of LCO 3.1.1.

3.1.4; Control Rod Scram Times:

In order for the “safe” endpoints credited in the Transient and Accident Analyses for the licensee a given scram insertion time had to be assumed. This specification recognizes that wear and age effect scram times and provides allowances while ensuring the endpoints described above are not sacrificed.

3.1.5; Control Rod Scram Accumulators:

Scram accumulator operability is essential to meeting the scram times of 3.1.4 but scram times are only evaluated periodically. Instrumentation on the scram accumulators monitors their performance continuously.

3.1.6; Rod Pattern Control:

Due to the limited number of LPRM strings with respect to fuel bundles, the concept of “pseudo strings” is employed in monitoring core performance. This concept uses actual LPRM data from one core location to assume the conditions in another core location that is absent an LPRM string. For this reason control rod symmetry is of the utmost importance. The goal of the banked position withdrawal sequence (BPWS) is establishing and maintaining control rod symmetry.

3.1.7; Standby Liquid Control (SLC) System:

Although not a replacement for the Safety Function of the Reactor Protection System, the Standby Liquid Control system provides a diverse independent means of accomplishing the control rod negative reactivity should the control rod insertion capability fail.

3.1.8; Scram Discharge Volume (SDV) Vent and Drain Valves:

These valves have dual safety functions. In the event of a scram, they close to prevent a leakage path from the RPV directly to secondary containment via the scram discharge volumes. When no scram is present, they are open to ensure sufficient volume remains in the scram discharge volume to accommodate a scram.

3.0.5.2 Power Distribution Limits (Section 3.2)

This section addresses the requirement to operate within the Thermal Limit operating limitations. The specific values of the limitations will be provided with the CORE OPERATING LIMITS REPORT (COLR) prior to startup from each refueling outage. The section includes:

3.2.1; AVERAGE PLANER LINEAR HEAT GENERATION RATE (APLHGR):

As defined in Section 1.1 the APLHGR is related to the LHGR for a given cross-sectional nodal elevation of a given bundle. Exceeding this limitation may result in post-LOCA peak clad temperatures above 2200°F.

3.2.2; MINUMUM CRITICAL POWER RATIO (MCPR):

As defined in Section 1.1, MCPR is a ratio of the power level at which a bundle may experience boiling transition to the actual power level of the bundle. This limitation also has a Safety Limit in Section 2.0. The operating limit of 3.2.2 accommodates the delta CPR challenge resulting from the analyses of RPV pressure transients. In either case the purpose of the limitation is to minimize the possibility of fuel failure due to boiling transition.

3.2.3; LINEAR HEAT GENERATION RATE (LHGR) (optional):

As defined in Section 1.1 LHGR is a measure of the heat generation rate per unit length of fuel rod, typically given in units of KW/ft. High generation rates can result in exceeding the plastic strain limitation of the fuel clad due to differential expansion contact with the fuel pellets. This limitation is shown as optional in that some COLR evaluation results reveal that the APLHGR limitation is more restrictive.

3.2.4; Average Power Range Monitor (APRM) Gain and Setpoints (optional):

Fundamental to the Transient and Accident Analyses is the assumption that the APRM's are operable and that their automatic function occurs conservative to actual conditions. At times of local power peaking, the APRM's may not adequately see this power and the local APLHGR and / or MCPR limits may be challenged. If this occurs, an adjustment is made to the APRM indication or automatic setpoints to compensate for this error.

3.0.5.3 Instrumentation (Section 3.3)

This section addresses the reactor plant parameter monitoring instrumentation and automatic setpoints. Many of the specifications contain tables and require a strong attention to detail. The section includes:

3.3.1.1; Reactor Protection System (RPS) Instrumentation:

This section contains all of the setpoint information relating to the automatic scram function. Many of these setpoints are Limiting Safety System Settings in that they initiate automatic corrective action (scram) before a Safety Limit can be violated. The majority of the information is in Table 3.3.1.1-1 and for each function, detailed information such as setpoint value or number of required instruments can be found. To properly use this section, you must first evaluate the impact of the problem against the Table then apply that to the correct Condition if non-compliance is detected.

3.3.1.2; Source Range Monitor (SRM) Instrumentation:

This section describes the operability requirements and applicable actions to be taken for the four SRM detectors located in the core. The setpoints for the SRM protective functions are located within the Technical Requirements Manual (TRM).

3.3.2.1; Control Rod Block Instrumentation:

Of all of the instrumentation that initiates Control Rod Blocks, only the Rod Block Monitor, Rod Worth Minimizer and Reactor Mode Switch met the LCO selection criterion discussed earlier. Also in tabular form, the specific setpoints are identified. Use of this specification is similar to the guidance of 3.3.1.1 above. The other instrumentation that initiates control rod blocks (NMS's) will be controlled by the Technical Requirements Manual (TRM)

3.3.2.2; Feedwater and Main Turbine High Water Level Trip Instrumentation:

With this instrumentation apparently protecting secondary plant equipment it could be asked which of the LCO criterion discussed earlier caused it to be selected. This instrumentation terminates the injection into the RPV and causes an indirect reactor scram for protection against the Feedwater Malfunction – Maximum Demand analyzed transient. This automatic action also terminates the power rise due to colder feedwater before MCPR is violated.

3.3.3.1; Post Accident Monitoring (PAM) Instrumentation:

Of all of the instrumentation found in a plant, only a small subset has the qualifications and range to monitor fission product barrier performance during an accident. The functions of these instruments (Table 3.3.3.1-1) are the topic of this specification. The specific instruments will be found in the Technical Requirements Manual.

3.3.3.2; Remote Shutdown System:

Although highly unlikely, issues such as toxic gas intrusion, fires or sabotage can challenge the ability of the operator to properly control the facility from the control room. The Remote Shutdown System provides the ability to shutdown the plant to MODE 3 from location(s) outside the control room in the event that control room accessibility and / or control are challenged. This specification is related to the OPERABILITY of operator functions and indications at remote locations as described in the BASES and Table B 3.3.3.2-1.

3.3.4.1; End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation:

When the plant is operating near the end of the operating cycle, control rod density is near zero meaning most if not all control rods are fully withdrawn. As compared to the beginning of cycle, when control rod density is higher, the result is that at the end of cycle, it takes somewhat longer to start and complete the negative reactivity effects of control rod insertion. This delay may adversely impact the analyzed pressure transient effect on the MCPR Thermal Limitation. For this reason, **some** plants include this instrumentation to expedite negative reactivity insertion by tripping the recirculation pumps directly from a sensed turbine trip or control valve fast closure.

3.3.4.2; Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation:

The majority of the transients affecting the reactor pressure vessel will be detected and terminated by a high reactor pressure or low reactor water level scram signal. Failure of the scram will result in further degradation of the condition and challenging the fuel and RPV integrities. The ATWS-RPT instrumentation detects this failure to scram condition and helps mitigate it by tripping the recirculation pumps.

3.3.5.1; Emergency Core Cooling System (ECCS) Instrumentation:

This section describes and places OPERABILITY controls on the instrumentation associated with ECCS subsystem initiation, operation and control. Similar to the RPS Instrumentation specification, the majority of the information is in Table 3.3.5.1-1 and for each function; detailed information such as setpoint value or number of required instruments can be found. To properly use this section, you must first evaluate the impact of the problem against the Table then apply to the correct Condition if non-compliance is detected.

3.3.5.2: Reactor Core Isolation Cooling (RCIC) Instrumentation:

This section describes and places OPERABILITY controls on the instrumentation associated with RCIC system initiation, operation and control. Similar to the RPS Instrumentation specification, the majority of the information is in Table 3.3.5.2-1 and for each function; detailed information such as setpoint value or number of required instruments can be found. To properly use this section, you must first evaluate the impact of the problem against the Table then apply it to the correct Condition if non-compliance is detected.

3.3.6.1; Primary Containment Isolation Instrumentation:

This section describes the controls on instrumentation associated with establishing primary containment integrity at various penetration flowpaths during accident conditions. This instrumentation is measuring process variables that result in a specific system isolation. It should be noted that some instruments process isolation signals to multiple systems (RPV water level) while others may only isolate the particular system (main steam line low pressure). Similar to the RPS Instrumentation specification, the majority of the information is in Table 3.3.6.1-1 and for each function; detailed information such as setpoint value or number of required instruments can be found. To properly use this section, you must first evaluate the impact of the problem against the Table then apply it to the correct Condition if non-compliance is detected.

3.3.6.2; Secondary Containment Isolation Instrumentation:

This section describes the controls on instrumentation associated with establishing secondary containment integrity at various penetration flowpaths during accident conditions. Similar to the RPS Instrumentation specification, the majority of the information is in Table 3.3.6.2-1 and for each function; detailed information such as setpoint value or number of required instruments can be found. To properly use this section, you must first evaluate the impact of the problem against the Table then apply it to the correct Condition if non-compliance is detected.

3.3.6.3; Low-Low Set (LLS) Instrumentation:

Multiple rapid operation of Safety Relief Valves places significant stresses on the suppression pool and the SRV tailpipes as well as raising the chances that the SRV may stick open. For these reasons, some plants have a low-low set function that once initiated, lowers the opening setpoint on certain SRVs to keep them open longer and reduce the frequency of valve operation. Similar to the RPS Instrumentation specification, the majority of the information is in Table 3.3.6.3-1 and for each function; detailed information such as setpoint value or number of required instruments can be found. To properly use this section, you must first evaluate the impact of the problem against the Table then apply it to the correct Condition if non-compliance is detected.

3.3.7.1; Main Control Room Environmental Control (MCREC) System Instrumentation:

During accident conditions, it may become necessary to protect the operators from radioactive release intrusion. Similar to the RPS Instrumentation specification, the majority of the information is in Table 3.3.7.1-1 and for each function; detailed information such as setpoint value or number of required instruments can be found. To properly use this section, you must first evaluate the impact of the problem against the Table then apply it to the correct Condition if non-compliance is detected.

3.3.8.1; Loss of Power (LOP) Instrumentation:

This section controls the instrumentation that starts the EDG's upon sensed degraded or lost voltage to the safety busses. Similar to the RPS Instrumentation specification, the majority of the information is in Table 3.3.8.1-1 and for each function; detailed information such as setpoint value or number of required instruments can be found. To properly use this section, you must first evaluate the impact of the problem against the Table then apply it to the correct Condition if non-compliance is detected.

3.3.8.2; Reactor Protection System (RPS) Electric Power Monitoring:

The reactor protection system components (relays and solenoids) are highly vulnerable to an unregulated power supply. Electrical protection assemblies in each of the power supplies provide isolation of these components from the power supply upon detection of under voltage, over voltage or under frequency.

3.0.5.4 Reactor Coolant System (Section 3.4)

This section addresses operational conditions related to the reactor coolant system. The section includes:

3.4.1; Recirculation Loops Operating:

Plant safety evaluations and some protective systems are based upon the normal condition of 2 loops in operation with matched flows. Should this condition not be met, action is required to ensure that the loss of core flow through the idle jet pumps does not result in a reduction in safety margins.

3.4.2; Jet Pumps:

Much of the plant safety analysis assumes that the jet pumps are intact and therefore the core can be reflooded, at least, to 2/3 core height (jet pump throat elevation).

3.4.3; Safety/Relief Valves (S/RV's):

The safety relief valves automatically open mechanically at their setpoint pressure to provide over pressure protection to the reactor coolant system pressure boundary. The setpoints are limiting safety system settings in that they relieve pressure to prevent violation of the RCS Pressure Safety Limit.

3.4.4; RCS Operational LEAKAGE:

LEAKAGE is categorized and defined in Section 1.1. This section places operational limitations on RCS leakage based upon the type, amount and source of the leakage.

3.4.5; RCS Pressure Isolation Valve (PIV) Leakage:

There are a number of systems with lower design pressures that interface with the reactor coolant system such as RHR. During normal power operations, these systems are isolated from the reactor coolant system by Pressure Isolation Valves (PIV's). If PIV leakage becomes excessive, the potential to overpressurize the low pressure system can result in an "interfacing system LOCA".

3.4.6; RCS Leakage Detection Instrumentation:

Remote monitoring of RCS leakage into containment is provided by several detection methods. This monitoring capability ensures compliance with LCO 3.4.4.

3.4.7; RCS Specific Activity:

RCS activity is measured periodically to ensure continued operation within the assumptions of the safety analysis. The analysis ensures that 10 CFR 100 Accident Release Rate Limitations are not violated provided that minimal coolant activity levels exist prior to the accident.

3.4.8; RHR Shutdown Cooling System – Hot Shutdown:

It is never acceptable to intentionally operate without forced circulation through the core to remove decay heat. Additionally, to operate the RHR system in the Shutdown Cooling Mode, you must disable the LPCI function at a time it is required to be OPERABLE. The specification notes and bases for 3.5.1 and 3.4.8 place conditions on allowing this to occur. This specification addresses both issues provided that Mode 3 operation is at a pressure below the SDC Cut-in Permissive Setpoint of 125 psig.

3.4.9; RHR Shutdown Cooling System – Cold Shutdown:

Essentially the same as LCO 3.4.8 except that while operating in Mode 4, the requirements of 3.4.9 and 3.5.2 provide more flexibility.

3.4.10; RCS Pressure and Temperature (P/T) Limits:

Operational limitations are placed upon RPV pressure, temperature and the rate of change of either primarily to minimize the potential for RPV brittle failure. The plant specific heatup and cooldown curves can be found in the Pressure Temperature Limits Report (PTLR). These limitations can be challenged during RPV heatup and cooldown as well as startup of an idle recirculation pump while hot. Additionally, these limitations help minimize the cold water introduction power excursion resulting from idle loop startup while at power.

3.4.11; Reactor Steam Dome Pressure:

To be conservative with the evaluations, all of the FSAR transient and accident analyses assume initial plant operating conditions that are beyond the normal operational conditions. This specification ensures plant operation remains within the assumed value of RPV pressure.

3.0.5.5 Emergency Core Cooling System (ECCS) and Reactor Core Isolation Cooling System (RCIC) (Section 3.5)

This section addresses the OPERABILITY requirements related to the ECCS and RCIC systems. The section includes:

3.5.1; ECCS – Operating:

The ECCS system is the primary system for ensuring that the ECCS Acceptance Criteria of 10CFR50.46 are met. The need for and challenges to the ECCS system are greater when the reactor is heated as in MODEs 1-3. Note that LCO 3.0.4.b is not applicable to HPCI meaning that a risk assessment cannot be used to justify a MODE change when HPCI is inoperable.

3.5.2; ECCS – Shutdown:

When the reactor is at lower energy states such as MODEs 4 and 5, there is less demand on the ECCS systems to meet the ECCS Acceptance Criteria. For this reason, requirements are lightened which allows for system inoperability to support outage maintenance activities.

3.5.3; RCIC System:

While operating at higher energy states (MODEs 1-3) RCIC is needed for inventory makeup in the event of a MSIV Isolation. Notice that LCO 3.0.4.b is not applicable to RCIC either.

3.0.5.6 Containment Systems (Section 3.6)

Provided the containment systems are operated and maintained within design conditions, they can accommodate the maximum temperature, pressure and radiological leakages for all of the Design Basis Accidents without exceeding the allowable Accident Release Limits of 10CFR-100. This section addresses the conditions that ensure the containment systems are operated within these design conditions. The section includes:

3.6.1.1; Primary Containment:

Primary Containment is considered OPERABLE if it is intact and can be isolated such that leakage does not exceed the maximum allowable total leakage rate at accident pressure. Specific conditions of OPERABILITY are contained within the Bases.

3.6.1.2; Primary Containment Air Lock:

Air locks allow primary containment access during periods when primary containment is required to be OPERABLE. An OPERABLE airlock meets allowable leakage criteria and is interlocked to prevent simultaneous opening of both doors. When the reactor is at low energy states, this requirement is relaxed to allow opening of both doors to support activities within primary containment.

3.6.1.3; Primary Containment Isolation Valves (PCIVs):

When primary containment is required to be OPERABLE, all process flow penetrations through the primary containment wall must be either isolated and leak tight or contain automatically closing primary containment isolation valves that close in a designated time interval and are also leak tight. This is a general specification to all penetrations while the specific penetration identification and valves are located in the Facility TRM.

3.6.1.4; Drywell Pressure:

The positive pressure maintained in Primary Containment is limited to a maximum value assumed in the safety analyses to ensure that the accident peak pressures do not exceed the containment design pressures. In Mark I and II Primary Containments, it is desirable to maintain a slight positive internal pressure to ensure that the inerted atmosphere is maintained and not diluted by oxygen intrusion.

3.6.1.5; Drywell Air Temperature:

Again, ensuring that plant operation is within the safety analysis assumed value of drywell air temperature prevents exceeding containment design temperature in the event of an accident.

3.6.1.6; Low-Low Set (LLS) Valves:

As discussed in 3.3.6.3 (LLS Instrumentation), multiple rapid operation of Safety Relief Valves places significant stresses on the suppression pool and the SRV tailpipes as well as raising the chances of a stuck open SRV. For these reasons, some plants have a low-low set function that automatically operates select valves on a wider pressure band to reduce the frequency of valve operation.

3.6.1.7; Reactor Building-to-Suppression Chamber Vacuum Breakers:

Mark I and Mark II plants that are susceptible to exceeding their negative design pressure have these valves. They serve a dual safety function. Normally they are closed to ensure containment integrity. Upon conditions of suppression chamber pressure less than reactor building pressure they open to prevent the negative design pressure of the primary containment. (Mark III has an equivalent vacuum breaker system discussed in the ITS for BWR-6 (NUREG-1434).

3.6.1.8; Suppression Chamber-to-Drywell Vacuum Breakers:

These valves are critical to containment accident response and also have a dual safety function. When drywell pressure exceeds suppression chamber pressure, they are closed to ensure a steam bypass flowpath to the suppression chamber airspace does not occur. When drywell pressure is below suppression chamber pressure, they open to vent non-condensibles back to the drywell to prevent exceeding drywell negative design pressure. (Mark III has an equivalent vacuum breaker system discussed in the ITS for BWR-6 (NUREG-1434).

3.6.1.9; Main Steam Isolation Valve (MSIV) Leakage Control System (LCS):

A Main Steam Line Break Accident that results in fuel damage, carries a large amount of fission products down the main steam lines before the MSIV's close. For those plants that have the LCS, it is in place to ensure that after an MSIV Isolation, the volume trapped between the MSIV's, the gland leakoff from the MSIV's and the downstream MSL piping is processed via a filtered (typically SGT) discharge path rather than allowed to vent directly to the Turbine Building/steam tunnel atmosphere.

3.6.2.1; Suppression Pool Average Temperature:

The DBA-LOCA analysis assumes an initial suppression pool temperature of 110 degrees F such that the energy released from a DBA-LOCA is completely condensed therefore ensuring primary containment does not exceed its design parameters. This specification ensures the normal operational margin to this assumed value.

3.6.2.2; Suppression Pool Water Level:

Related to 3.6.2.1, peak accident containment pressure and temperature is directly related to the suppression pool water volume. Too little volume can result in incomplete condensation of the released steam resulting in excessive primary containment pressurization. Although good for condensing ability, too much water volume means too little gas volume. The non-condensables must now exist in a smaller space resulting in higher peak containment pressures.

3.6.2.3; Residual Heat Removal (RHR) Suppression Pool Cooling:

Post-LOCA, startup of the suppression pool cooling system is the first safety analysis action required to be performed by the operating staff (10 minutes after the LOCA). This action is required to terminate continued containment heatup and pressurization resulting from the long term decay heat rejected from the nuclear core.

3.6.2.4; Residual Heat Removal (RHR) Suppression Pool Spray:

Again, to prevent exceeding primary containment design parameters the suppression pool spray mode is used to reject heat from the suppression chamber airspace. Ideally, it should not be needed as suppression pool cooling should be adequate. Realistically some minor leakage across the suppression chamber to drywell vacuum breakers will result in minor amounts of steam in the suppression chamber air space. Suppression Pool Spray will condense this steam minimizing the pressure impact. From a single failure perspective, if one of these vacuum breakers were fully open, a large volume of steam collects in the suppression chamber airspace and in this case suppression pool spray is REQUIRED to protect containment from failure.

3.6.2.5; Drywell-to-Suppression Chamber Differential Pressure:

In Mark I plants, this D/P specification keeps drywell pressure above suppression chamber pressure such that the water level inside the downcomers is significantly lower than outside which reduces the hydraulic stress of a blowdown on the primary containment. Specifically, this D/P minimizes the peak drywell pressure resulting from a DBA-LOCA, minimizes the hydrodynamic forces on the torus (lift potential) and minimizes the fluid hammer shock to the 96 downcomer pipes.

3.6.3.1; [Drywell Cooling System Fans]:

At some plants, the function of the drywell cooling system fans is safety related in that it maintains the post-accident atmosphere mixed to prevent hydrogen pocketing and the potential for a related fire or explosion.

3.6.3.2; Primary Containment Oxygen Concentration:

Hydrogen at 4-78% in an oxygenated atmosphere is flammable to explosive. During power operation the primary containment atmosphere is inerted with nitrogen to establish and maintain a very low concentration of oxygen. This ensures that any hydrogen produced as a result of the accident is not exposed to an oxygenated atmosphere and is therefore not flammable or explosive.

3.6.3.3; Containment Atmosphere Dilution (CAD) System:

Post-accident, it is desirable to maintain the inerted drywell atmosphere while providing the capability to feed and bleed to remove hydrogen. The CAD system is a large volume of nitrogen with the required components to accomplish this purge.

3.6.4.1; [Secondary] Containment:

The secondary containment function is important to both the DBA-LOCA and the DBA-Fuel Handling accidents. The secondary containment (typically the reactor building) provides a holdup and dilution volume for post-accident radiological releases from primary containment or dropped fuel bundle releases. It operates in conjunction with the secondary containment isolation valves and the standby gas treatment system to ensure the secondary containment volume is processed via a filtered elevated release point rather than building leakage as a ground level release.

3.6.4.2; Secondary Containment Isolation Valves (SCIVs):

These valves ensure that process flow penetrations through the secondary containment boundary are isolated during the above accident conditions thus preventing ground level release pathways to the environment.

3.6.4.3; Standby Gas Treatment (SGT) System:

The SGT system is the filtered elevated release pathway of the secondary containment volume. Its functions are to maintain the secondary containment at a slightly negative pressure which prevents ground level releases and to filter the discharged volume to minimize the radioactive releases before discharge to the elevated release point.

3.0.5.7 Plant Systems (Section 3.7)

This section is a collection of various safety significant systems that are not located elsewhere in the specifications. In many cases, OPERABILITY of these support systems affect OPERABILITY of supported systems. With this in mind, these specifications frequently find relief in a required safety function determination as stated in LCO 3.0.6. The section includes:

3.7.1; Residual Heat Removal Service Water (RHRSW) System:

The primary mission of the RHRSW system is connection to the ultimate heat sink for heat rejection from the RPV via the RHR system. This could be while operating in the normal configuration of the shutdown cooling mode or in the accident configuration of suppression pool cooling. Depending upon the plant, this function may be served by an independent system, a sub function of another system (SW), and/or in combination with another function (EDG-SW).

3.7.2; Plant Service Water (PSW) System and Ultimate Heat Sink(UHS):

This system provides connection to the ultimate heat sink for various safety related loads such as RHR pump coolers or room coolers where safety related equipment is located. As described above, this function may be combined in the same system with other safety related heat sink functions.

3.7.3; Diesel Generator (DG) Standby Service Water (SSW) System:

For continued operation, the heat produced by the diesel generator must be rejected to the ultimate heat sink. This system provides that heat rejection path. Again, as described above, this function may be combined in the same system with other safety related heat sink functions.

3.7.4; Main Control Room Environmental Control (MCREC) System:

During accident conditions, it may become necessary to protect the operators from radioactive release intrusion. This system will automatically enter a recirculation mode with a filtered makeup air supply to protect the operators. Ultimately it is based upon limiting the operator dose to less than 5 rem whole body over a 30 day exposure period. This system may operate in conjunction with a toxic gas monitoring system at those plants with a potential for toxic gas intrusion. An example is a plant near a railroad where toxic substances such as chlorine or ammonia are transported.

3.7.5; Control Room Air Conditioning (AC) System:

In the event of MCREC system initiation personnel as well as the equipment in the control room are heat loads that will result in elevated temperatures. This system provides a heat sink to the environment maintaining a controlled temperature and humidity environment. This ensures continued OPERABILITY of the control room components.

3.7.6; Main Condenser Offgas:

During normal plant operation, the steam jet air ejectors evacuate large volumes of water vapor as well as volatile radioactive gasses, such as isotopes of Xe, Kr, O and H. To minimize the radioactive releases to the environment, the offgas system processes these releases, recombining the H and O, condensing the resulting vapor for return to the main condenser, holdup of flow for radioactive decay and filtering the remaining volume before release. Maintaining the gross gamma concentration low minimizes the potential releases in the event of an offgas system pressure boundary failure.

3.7.7; Main Turbine Bypass System:

The turbine bypass valves function during a turbine trip to minimize the pressure transient on the RPV and therefore the resulting challenge to fuel thermal limits. If the turbine bypass system is not fully capable of performing this function then fuel thermal limit penalties are imposed.

3.7.8; Spent Fuel Storage Pool Water Level:

This specification ensures that in the event of a fuel handling accident (dropped fuel bundle over the SFP) sufficient water volume is present to absorb and retard fission product release to the secondary containment atmosphere and ultimately to the public.

3.0.5.8 Electrical Power Systems (Section 3.8)

This section addresses the electrical power sources and distribution systems that ensure reliable power to safety significant station equipment. In many cases, OPERABILITY of these support systems affects OPERABILITY of supported systems. With this in mind, the supported system specifications frequently find relief in a required safety function determination as stated in LCO 3.0.6. The section includes:

3.8.1; AC Sources – Operating:

With the reactor at higher energy states there is a need for maximum safety related equipment and therefore a need for maximum reliable power sources. The preferred power source is the dedicated off-site distribution network, backed up by the on-site emergency diesel generators. The automatic sequencers provide for time delayed starting of large components (RHR/CS pumps) protecting the diesel generators from overload as well as preventing a bus loss caused by supply breaker overcurrent. Notice that LCO 3.0.4.b risk evaluations are not applicable to the diesel generators.

3.8.2; AC Sources – Shutdown:

Conversely with the above, at lower energy states, there is less safety equipment demand and therefore lighter demands on the required power sources.

3.8.3; Diesel Fuel Oil, Lube Oil and Starting Air:

These systems support initial start and continuous operation of the station Emergency Diesel Generators. The air starting system provides for several (5) starting attempts without recharging. The fuel oil and the lube oil systems provide for 7 continuous days of operation.

3.8.4; DC Sources – Operating:

Much of the safety significant equipment relies upon battery power for logic and control power. Some systems (HPCI/RCIC) rely upon battery power for component operation. The batteries are sized to provide a reliable source of power for these functions in the event of a loss of the battery chargers during accident conditions. For these applications the batteries are sized to support full component load operation for a site specific amount of time based upon the safety analysis. This loading time is less than and completely independent of the station blackout (SBO) rule coping time required by 10CFR50.63.

3.8.5; DC Sources – Shutdown:

In a shutdown condition, the safety significant equipment OPERABILITY requirements are relaxed therefore the need for multiple divisions of reliable DC power are also relaxed.

3.8.6; Battery Parameters:

Battery OPERABILITY is conditioned upon monitoring and maintaining acceptable battery parameter values designated by Specification 5.5.14, IEEE Standard 450-1995. Parameters of note are cell voltage, cell temperature, cell electrolyte level, battery float current, and battery capacity.

3.8.7; Inverters – Operating:

At some plants, reliable AC power is supplied to the reactor protection (RPS) and emergency core cooling (ECCS) system logics from AC Inverters. The power supply may be safety related AC through a rectifier to an internal battery or directly from the station battery. This DC power is then inverted to AC to provide power to the logics.

3.8.8; Inverters – Shutdown:

In a shutdown condition, the safety significant equipment OPERABILITY requirements are relaxed therefore the need for multiple inverter divisions are also relaxed.

3.8.9; Distribution Systems – Operating:

Per the earlier 3.8 specifications, OPERABILITY requirements are set forth for AC, DC and inverter power sources in plant operating conditions. This specification ensures a reliable means of supplying that required power to the appropriate components.

3.8.10; Distribution Systems – Operating and Shutdown:

In a shutdown condition, the safety significant equipment OPERABILITY requirements are relaxed therefore the need for multiple buses to support reliable power distribution are also relaxed.

3.0.5.9 Refueling Operations (Section 3.9)

This section addresses the unique conditions related to plant operation in MODE 5; Refueling. The section includes:

3.9.1; Refuel Equipment Interlocks:

Operation in this MODE has the potential for inadvertent criticalities as well as challenges to bridge personnel safety. If a control rod is being withdrawn at the same time that an adjacent bundle is being lifted out of the core, the **EFFECT** is much higher rod speed and therefore a much higher reactivity addition rate. Besides the possibility of fuel damage, the safety of the personnel on the refuel platform over the core is compromised. The refueling interlocks limit operations to any 2 of the 3 following items:

- Bridge travel over the core
- Control rod withdrawal
- Any bridge hoist loaded with fuel

3.9.2; Refuel Position One-Rod-Out Interlock:

With the reactor MODE switch in the Refuel position, interlocks prevent the withdrawal of any control rod if any other control rod in NOT fully inserted. Restated, no more than 1 control rod can be withdrawn from full in. This interlock prevents the withdrawal of control rods, which in the right circumstances can result in an inadvertent reactor criticality.

3.9.3; Control Rod Position:

Generally speaking, during refueling activities when core reactivity is increasing as fuel is loaded, all control rods must be fully inserted to ensure that adequate shutdown margin is maintained at all times. There are specific exceptions to this requirement in Section 3.10.

3.9.4; Control Rod Position Indication:

While operating in MODE 5, the refueling interlocks in 3.9.1 are required to be OPERABLE. Some of these interlocks receive input from the control rod full in position indicators which must therefore also be OPERABLE. Again, there are specific exceptions to this requirement in Section 3.10.

3.9.5; Control Rod OPERABILITY – Refueling:

Although refueling interlocks should prevent this, there still exists the potential to insert fuel into a location that has a fully withdrawn control rod. This could result in an inadvertent criticality. Therefore, in a refueling condition any fully withdrawn control rod must be operable including its scram capability.

3.9.6; Reactor Pressure Vessel (RPV) Water Level – Irradiated Fuel:

Similar to the Spent Fuel Pool Level requirement of 3.7.8, this specification ensures that in the event of a fuel handling accident (dropped bundle over the RPV) sufficient water volume is present to absorb and retard fission product release to the secondary containment atmosphere and ultimately to the public.

3.9.7; Reactor Pressure Vessel (RPV) Water Level – New Fuel or Control Rods:

This condition allows for a much lower water level (volume), compared to specification 3.9.6 because only ½ of the collision from a dropped fuel bundle or control rod involves irradiated fuel.

3.9.8; Residual Heat Removal (RHR) – High Water Level:

The added water volume of a flooded reactor cavity provides for a larger heat sink for natural circulation cooling of core decay heat. In MODE 5 with this volume of water in the core maintained, this specification substitutes for LCO 3.4.9's intent to have TWO OPERABLE Shutdown Cooling subsystems.

3.9.9; Residual Heat Removal (RHR) – Low Water Level:

Compared to 3.9.8 above, if the volume of water in the core is not maintained then one additional loop of SDC is required to be OPERABLE in the event that the operating loop is lost.

3.0.5.10 Special Operations (Section 3.10)

As the name implies, this section deals with special circumstances and enforces special conditions to accommodate them. The section includes:

3.10.1; Inservice Leak and Hydrostatic Testing:

RPV pressure testing is performed at the end of the refueling outage to demonstrate RPV integrity before any MODE change from MODE 4. The test itself results in a plant condition of Table 1.1-1 that is MODE 3 because of the elevated temperatures required by the PTLR curve for non-critical operation. In this condition, much of the equipment required to be operable (primary containment and ECCS) in MODE 3 will not be OPERABLE for a variety of reasons. This specification provides relief from declaring the MODE changes if the special conditions are satisfied.

3.10.2; Reactor Mode Switch Interlock Testing:

Reactor mode switch interlock testing permits positioning of the mode switch such that interlock testing may be performed to demonstrate operability of interlocks before the actual MODE requirement for OPERABILITY, such as SRMs and IRMs Requiring OPERABILITY of other equipment at this point is unreasonable or unnecessary. This specification provides relief from declaring the MODE changes if the special conditions are satisfied.

3.10.3; Single Control Rod Withdrawal – Hot Shutdown:

As defined in Table 1.1-1, MODE 3 has the reactor mode switch in the shutdown position which results in a permanent control rod block. In this condition, it may become necessary to withdraw a control rod to support any of a number of testing activities. To do so requires placing the MODE switch in refuel which becomes MODE 2 and requires a number of other OPERABILITY requirements that may not be desirable or necessary. This specification provides relief from declaring the MODE changes if the special conditions are satisfied.

3.10.4; Single Control Rod Withdrawal – Cold Shutdown:

As defined in Table 1.1-1, MODE 4 has the reactor mode switch in the shutdown position which results in a permanent control rod withdrawal block. In this condition, it may become necessary to withdraw a control rod to support any of a number of testing activities. To do so requires placing the MODE switch in refuel which becomes MODE 2 and requires a number of other OPERABILITY requirements that may not be desirable or necessary. This specification provides relief from declaring the MODE changes if the special conditions are satisfied.

3.10.5; Single Control Rod Drive (CRD) Removal – Refueling:

While operating in MODE 5, if it becomes necessary to replace a control rod drive mechanism in a core cell containing fuel, a number of obstacles present themselves. A disconnected drive means the scram function is inoperable for that rod. Removal of the PIP probe to support CRD replacement impacts the single rod removal interlock. This specification provides relief from those OPERABILITY requirements for that control rod only if the special conditions are satisfied.

3.10.6; Multiple Control Rod Withdrawal – Refueling:

To support various activities (inspections, control rod blade replacement) during a refueling outage, it may be necessary to empty a number of fuel cells. This means all 4 fuel assemblies removed and the control rod withdrawn or removed. With the reactor MODE switch in the refuel position, interlocks prevent the withdrawal of more than one control rod. If there is no fuel in the fuel cell, the control rod worth is very low and it has little influence on core SHUTDOWN MARGIN. This specification provides relief from those interlocks and other OPERABILITY requirements for multiple control rods only if the special conditions are satisfied.

3.10.7; Control Rod Testing – Operating:

LCO 3.1.6 requires the control rod pattern to comply with the banked position withdrawal sequence (BPWS). This means that all control rods in a given group are maintained within 1 notch position of each other. At times, testing is required (SDM, scram timing) that results in violation of the 3.1.6 requirement and the resulting control rod blocks from the Rod Worth Minimizer (RWM). This specification provides relief from the RWM and BPWS OPERABILITY requirements if the special conditions are satisfied.

3.10.8; SHUTDOWN MARGIN (SDM) test – Refueling:

The SHUTDOWN MARGIN test requires the withdrawal of multiple control rods while in MODE 5. Interlocks prevent withdrawing more than 1 control rod with the reactor mode switch in the Refuel position. To allow multiple control rod withdrawal requires positioning the reactor MODE switch to Startup, such that a new Table 1.1-1 MODE 2 would be entered. This may require OPERABILITY of other equipment that at this point is unreasonable or unnecessary. This specification provides relief from declaring the MODE change if the special conditions are satisfied.

3.10.9; Recirculation Loops – Testing:

This specification provides relief from the requirements of LCO 3.4.1 and therefore allows power operation in the natural circulation MODE if the special conditions are satisfied. The safety analysis demonstrates adequate heat removal in the natural circulation mode to support up to 50% core thermal power.

3.10.10; Training Startups:

Most of this training is conducted on plant specific simulators. Although extremely infrequent in use, this special condition LCO allows relief from the LCO 3.5.1 LPCI OPERABILITY requirement for one loop while in MODE 2 conducting startup training. The startup and ascent to heating range multiple times results in large volumes of water discharge to the radwaste systems. This specification provides relief by allowing operation of one loop of RHR in the SDC mode to limit the temperature rise and therefore limit the discharge to radwaste.

Table 3.0-1 Custom Technical Specifications

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.1.5 SOLENOID-ACTUATED PRESSURE RELIEF VALVES (AUTOMATIC DEPRESSURIZATION SYSTEM)	3.1.5 SOLENOID-ACTUATED PRESSURE RELIEF VALVES (AUTOMATIC DEPRESSURIZATION SYSTEM)
Applicability:	Applicability:
Applies to the operational status of the solenoid actuated relief valves.	Applies to the operational status of the solenoid actuated relief valves.
Objective:	Objective:
To assure the capability of the solenoid-actuated pressure relief valves to provide a means of depressurizing the reactor in the event of a small line break to allow full flow of the core spray system.	To assure the operability of the solenoid-actuated pressure relief valves to perform their intended functions.
Specification:	Specification:
	The solenoid-actuated pressure relief valve surveillance shall be performed as indicated below.
a. During power operating condition whenever the reactor coolant pressure is greater than 110 psig and the reactor coolant temperature is greater than saturation temperature, all six solenoid-actuated pressure relief valves shall be operable.	a. At least once during each operating cycle, verify each valve actuator strokes when manually actuated.
b. If specification 3.1.5a above is not met, the reactor coolant pressure and the reactor coolant temperature shall be reduced to 110 psig or less and saturation temperature or less, respectively, within ten hours.	b. At least once during each operating cycle, automatic initiation shall be demonstrated.

Table 3.0-2 Standard Technical Specifications

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. 0.38% Δk/k with the highest worth rod analytically determined,
or

b. 0.28% Mk/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.
b. In OPERATIONAL CONDITION .3 or 4, immediately verify all insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.
c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS-and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle:

a. By measurement, prior to or during the first startup after each refueling.
b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.
c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

Table 3.0-3 Improved Standard Technical Specifications

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Reactor Steam Dome Pressure

LCO 3.4.11 The reactor steam dome pressure shall be < [1020] psig.

APPLICABILITY: MODES I and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	Verify reactor steam dome pressure shall be < [1020] psig.	12 hours

Table 3.0-4 Changes in End States

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Residual Heat Removal (RHR) Suppression Pool Spray

LCO 3.6.2.4 Two RHR suppression pool spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool spray subsystem inoperable.	A.1 Restore RHR suppression pool spray subsystem to OPERABLE status.	7 days
B. Two RHR suppression pool spray subsystems inoperable.	B.1 Restore one RHR suppression pool spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 ---------- NOTE ---------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------	
	Be in MODE 3.	12 hours
	~~AND~~	
	~~C.2 Be in MODE 4.~~	~~36 hours~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.4.1	Verify each RHR suppression pool spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise	31 days

Table 3.0-5 Risk-Informed Completion Time Extension

3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

LCO 3.6.2.3 Two RHR suppression pool cooling subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool cooling subsystem inoperable.	A.1 Restore RHR suppression pool cooling subsystem to OPERABLE status.	7 days [OR In accordance with the Risk Informed Completion Time Program]
B. Two RHR suppression pool cooling subsystems inoperable.	B.1 Restore one RHR suppression pool cooling subsystem to OPERABLE status.	8 hours [OR In accordance with the Risk Informed Completion Time Program]
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

Table 3.0-5 Risk-Informed Completion Time Extension (cont.)

[5.5.15 Risk Informed Completion Time Program

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODE 1, 2 [, and MODE 3 while relying on the main condenser for heat removal];

c. When a RICT is being used, any plant configuration change within the scope of the Risk Informed Completion Time Program must be considered for the effect on the RICT.

   1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

   2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

   3. Revising the RICT is not required If the plant configuration change would lower plant risk and would result in a longer RICT.

d. Use of a RICT is not permitted for voluntary entry into a configuration which represents a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE.

e. Use of a RICT is permitted for emergent conditions which represent a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE if one or more of the trains are considered "PRA functional" as defined in Section 2.3.1 of NEI 06-09.]

Table 3.0-6 Relocation of SR Frequency Outside TS

3.6 CONTAINMENT SYSTEMS

3.6.2.2 Suppression Pool Water Level

LCO 3.6.2.2 Suppression pool water level shall be ≥ [12 ft 2 inches] and ≤ [12 ft 6 inches].

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Suppression pool water level not within limits.	A.1 Restore suppression pool water level to within limits.	2 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.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.2.1	Verify suppression pool water level is within limits.	[24 hours <u>OR</u> In accordance with the Surveillance Frequency Control Program]

Table 3.0-6 Relocation of SR Frequency Outside TS (cont)

5.5.17 [Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.]

Table 3.0-7 Inoperable Barrier Impact on Supported Systems

LCO 3.0.9

When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

For the purposes of this specification, the High Pressure Core Spray system, the Reactor Core Isolation Cooling system, and the Automatic Depressurization System are considered independent subsystems of a single system.

If the required OPERABLE train or subsystem becomes inoperable while this specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.

ATTACHMENT 1 Suggested Approach in Addressing TS Problems

This attachment is intended as an aid in developing the skill set and discipline necessary to correctly resolve a technical specification application problem.

FOR ANY T/S RELATED PROBLEM:

- **Review your system understanding**
  - Of course, critical to this guidance is a sound system understanding.

- **Review TOC for potential LCO(s)**
  - Frequently a given component or issue may have effect on multiple specifications.

- **Review applicability of LCO(s)**
  - It is not infrequent that improper determinations are made due to the plant condition applicability

- **Review LCO bases**
  - A given component having effect on multiple specifications or with different requirements in multiple applicabilities may result in inoperability determinations in some but not necessarily all. The LCO bases will clarify these conditions.

- **Review SR's and bases**
  - A given surveillance that fails, cannot be performed or cannot be met does not necessarily result in equipment inoperability.

- **Apply definitions (ALL CAPS)**
  - Remember that ALL CAPS means some additional detail is contained in the definitions of section 1.1

- **Apply T/S motherhood statements**
  - The LCO 3.0 and SR 3.0 motherhood statements frequently provide twists to what seems to be black and white.

- **Identify the required actions**
  - What we are trying to accomplish.

ATTACHMENT 2 Sample Problems

Section 1.0 Sample Problems:

1. A pressure instrument is adjusted such that for each simulated pressure input, the current output is in the required range and accuracy. This is an example of which of the following:
   - CHANNEL CALIBRATION
   - CHANNEL CHECK
   - CHANNEL FUNCTIONAL TEST
   - LOGIC SYSTEM FUNCTIONAL TEST

2. With the unit at 20% RTP, a reactor trip occurs and the Mode Switch is placed in Shutdown. What MODE was the plant in before and after the trip?

3. While in MODE 1, an ADS valve becomes inoperable. What condition(s) in TS 3.5.1 must be entered? An hour after the original ADS valve failure, a second ADS valve is found to be inoperable. What condition(s) in TS 3.5.1 should be entered? An hour later, HPCI becomes inoperable. With HPCI and 2 ADS valves inoperable, what condition(s) in TS 3.5.1 should the licensee be in?

4. Does a Surveillance have to be performed to determine if a Surveillance Requirement is not met?

5. SR 3.1.4.2 requires a “representative Sample” of control rods to be tested. What constitutes a representative sample?

Section 2.0 Sample Problems:

1. Which two TS required functions help to ensure SL 2.1.1.1 is not violated? (Hint: Search the TS Bases document with “2.1.1.1”)

2. The RCS design pressure is 1250 psig. What is the basis for establishing the Safety Limit at 1325 psig?

Section 3.0 Sample Problems:

1. The “A” diesel is declared inoperable and Condition B of TS 3.8.1 is entered. If the diesel is repaired and tested in 8 hours, before either Required Action B.3.1or B.3.2 is completed, must one of them be completed in order to exit Condition B?

2. When an LCO is not met, LCO 3.0.4 states that entry into a mode or condition in the Applicability can only be made under specific circumstances. Can the unit

change from MODE 4 to Mode 2 if:

a. One of the 2 required Post-Accident Monitoring Vessel Level Instruments is not OPERABLE?

b. One train of SLC is not OPERABLE?

c. RCIC is not OPERABLE?

d. RCS Specific Activity is 3.0 µCi/gm DOSE EQUIVALENT I-131 with the MSIVs open?

Section SR 3.0 Sample Problems:

1. SR 3.8.1.1 is used to help assess the OPERABILITY of off-site power sources. If it is discovered the SR was last completed 8 days ago, is the LCO met? What if it was 12 days ago?

2. LCO 3.8.1.1, Condition A is entered when a required off-site AC power source is declared inoperable and ACTIONS A.1, A.2 and A.3 are entered. The next day the shift discovers that SR 3.8.1.1 was last completed 12 hours ago. What are the Required Actions?

Section 4.0 Sample Problems:

1. What is the maximum number of fuel assemblies that can be loaded in the Spent Fuel Pool?

2. How many fuel assemblies and control rods are contained in the core?

3. What is the maximum effective multiplication factor for fuel stored in the spent fuel pool?

Section 5.0 Sample Problems:

1. When must a Radiation Protection Technician be on site?

2. When can a change to the TS Bases be made without NRC approval?

Introduction Lecture Sample Problem:

1. The plant 20% RTP when an Over travel Alarm is received when a control rod is fully withdrawn. All other rods are OPERABLE. What are the Required Action(s)?

ATTACHMENT 2 Sample Problems

Section 3.1 Sample Problems:

1. After a refueling outage, a reactor startup and heatup is in progress and the reactor engineer delivers the result of the post-criticality SR 3.1.1.1 two hours after criticality. The calculated SHUTDOWN MARGIN is 0.29% Δk/k. In this calculation the highest worth control rod was determined by testing prior to startup. Is the SHUTDOWN MARGIN satisfactory for the current plant conditions?

2. While operating at full power, several periodic performances of SR 3.1.4.2 have resulted in declaring a total of 9 operable control rods “slow”, but none are inoperable. During today’s performance of SR 3.1.4.2, another operable control rod is declared “slow” and is directly adjacent to one of the previously declared “slow” rods. No other “slow” rods are in adjacent locations. What are the Required Actions for this situation?

3. With a reactor startup in progress at 2% power and steam dome pressure at 910 psig, an accumulator trouble alarm on a control rod is received and locally the accumulator pressure is found to be 0 psig. The accumulator cannot be re-pressurized. This control rod scram time has previously been declared “slow” per LCO 3.1.4. All other control rod accumulators are at their normal pressure. What are the Required Actions for this situation?

4. During SLC pump testing in MODE 1, the operator manually closes the pump suction valve from the Boron Tank and the pump suction valve from the Test tank is opened. What TS Action needs to be entered during the testing??

5. The reactor is operating at full power with one of the scram discharge volume drain valves inoperable. SR 3.1.8.2 is performed and one of the scram discharge volume vent valves fails to close. What are the Required Actions for this situation? What if it was the other drain valve that failed?

Section 3.2 Sample Problems:

1. With the reactor at 20% RTP, the reactor engineer reports that the MCPR values of SR 3.2.2.1 are exceeding the limitation of the COLR for MCPR at 100% RTP. What are the Required Actions for this situation?

2. While operating at full power, an operator performing SR 3.2.2.1 reports that the limiting MCPR core value is 1.06 while the COLR limitation is 1.40. What are the Required Actions for this situation?

ATTACHMENT 2 Sample Problems

Section 3.3 Sample Problems:

1. While in MODE 1, the “A” and “D” APRMs both fail. Is the LCO for the APRM trip function met? What if it was the “A” and “C” channels that had failed?

2. While in MODE 1, the “A” and “C” APRM channels fail low and the “A” channel is placed in a tripped condition. Can the unit continue to operate in this Condition for an unlimited period of time? Can the “A” channel remain in a tripped condition for an unlimited period of time?

3. While in MODE 1, one of the two required Drywell Pressure Post-Accident Monitoring instruments fails its surveillance. What is the required action?

4. While in MODE 1, the initiation timer for the ADS channel “A” fails. Assuming the LCOs are met for all other ECCS systems, how long can the timer remain inoperable before the ADS valves have to be declared inoperable?

5. While in MODE 1 with the RCIC system in a normal alignment, both the required CST level switches that cause an automatic swap of the RCIC suction to the suppression pool are declared inoperable. What is the required action?

6. While in MODE 1, one of the pressure switches in the main steam header that cause a Group 1 isolation, fails high. What action is required and what is the associated completion time? How long (from the time the failure was discovered) can the unit remain in MODE 1 without repairing the channel or placing it in trip?

Section 3.4 Sample Problems:

1. The plant is in MODE 1 with both recirculation pumps running. Total core flow is at 60 Mlbm/hr (≈78% rated core flow) with “A” loop flow at 28 Mlbm/hr (≈36% of rated flow) and “B” loop flow at 32 Mlbm/hr (≈41.5% of rated flow). Is LCO 3.4.1 met for two recirculation loops with matched flow? If not, which loop is inoperable?

2. Following an outage, the SRVs are being cycled per SR 3.4.3.2. During the testing, the solenoid valve on one SRV that allows the valve to be open with the control switch is found to be shorted out and blowing fuses. Is the LCO met if SR 3.4.3.2 cannot be completed due to the faulty solenoid?

3. The plant is in MODE 4 making preparations for reactor startup. RHR Pumps “B” and “D” are operable and the “B” pump is operating in shutdown cooling mode. There are no recirculation pumps running and both the “A” and “C” RHR pumps are under a clearance and cannot be operated. Is LCO 3.4.9 met for this condition?

Section 3.5 Sample Problems:

1. The plant is in MODE 1. RHR Pump "B" is removed from service for maintenance. Is LCO 3.5.1 met for this condition?

2. Following an outage, the SRVs are being cycled per SR 3.4.3.2. During the testing, the solenoid valve on one SRV that allows the valve to be open with the control switch is found to be shorted out and blowing fuses. If this SRV is also one of the 7 ADS valves, what action is required?

Section 3.6 Sample Problems:

1. The plant is in MODE 1, the Outboard RCIC Steam Line Warmup Isolation Valve, MOV E51-048, fails the quarterly stroke timing test (SR 3.6.1.3.6) and was left in the closed position following the failure. Which LCOs are not met and what are the Required Action(s)?

2. During a HPCI pump surveillance, suppression pool temperature reaches 100°F. Is LCO 3.6.2.1 met, and if not what is the Required Action(s)? Once the testing is completed, is LCO 3.6.2.1 met and if not what is the Required Action(s)?

3. What is required for an OPERABLE Suppression Pool Cooling subsystem? What is required for an OPERABLE Suppression Pool Spray subsystem?

Section 3.7 Sample Problems:

1. With the plant in MODE 1, the "A" RB Service Water pump fails the quarterly surveillance (use RHRSW LCO 3.7.1) and is declared inoperable. Is LCO 3.7.1 met and if not what is the Required Action?

2. During the valve stroke of SR 3.7.7.1 while at rated thermal power, a turbine bypass valve will not open. If the valve cannot be repaired, what actions can the licensee take to avoid reducing power?

3. While in MODE 1, a turbine bypass valve fails open, other than LCO 3.7.7, what other LCOs are impacted by this configuration? (Hint: search the Bases document using "turbine bypass valve").

ATTACHMENT 2 Sample Problems

Section 3.8 Sample Problems:

1. While in MODE 1, the electrical system dispatcher informs the utility that due to grid instability, one of the two off-site sources is no longer qualified to meet the post-accident frequency and voltage requirements. What are the Required Actions in LCO 3.8.1?

2. While in MODE 1, the Division I emergency diesel would not start when attempting to perform the monthly surveillance. The cause of the failure is not readily apparent. What are the required actions?

3. While in MODE 1 with the Division I emergency diesel inoperable, SR 3.8.1.1 fails for the Division II required off-site power source. Which Required Actions should the plant be in.

4. While in MODE 1, a diesel fuel oil tank is discovered to have only 30,000 gallons of fuel, but the FSAR requires 33,000 gallons to complete the mission time of 7 days. Is the diesel OPERABLE? What if the tank had only 27,000 gallons?

5. While in MODE 1, a diesel trouble alarm is received and the air start receiver is found to be at 100 psig. What are the required Actions?

Section 3.9 Sample Problems:

1. During refueling operations, the “full-in” rod position indication for a fully inserted control rod is lost. Can fuel movement continue?

2. The unit has just completed refueling operations and the level is > 23 feet above the RPV flange. The “A” and “C” RHR pumps are OPERABLE and the “A” pump is in operation in shutdown cooling mode. No other RHR pumps are OPERABLE and there is no alternate decay heat removal (ADHR) method available. Can the licensee lower level to < 23 feet above the flange?

Section 3.10 Sample Problems:

1. During Reactor Mode Switch Interlock Testing while in MODE 4, the Mode Switch is placed in Run. What operating mode is the unit now in? What additional plant restrictions need to be implemented during the testing?