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# ATTACHMENTS



# LIST OF FIGURES

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## 3.1.0 Analysis of Technical Specifications – Unit 1

**Learning Objectives:**

1. State the requirements for and briefly describe the categories included in technical specifications.
2. Demonstrate understanding of the meanings of all defined terms in the technical specifications by applying them correctly in operational scenarios.
3. Explain the significance of the safety limits and the limiting safety system settings.
4. When given an initial set of operating conditions, use the format and content of the technical specifications to identify the applicable section from which to determine the appropriate plant and/or operator response.

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### 3.1.1 Introduction

This section is the first of four technical specification sections. It briefly discusses the requirements for technical specifications, covers the technical specification areas of definitions and safety limits, and provides general guidance for the use and application of the limiting conditions for operation (LCOs) and surveillance requirements. Safety limits are limits upon plant parameters, such as pressure, power, temperature, and flow, which are necessary to prevent fuel damage or the release of radioactive material. Limiting conditions for operation identify the required performance levels for safe operation, in terms of equipment operability and limits on certain plant parameters in areas such as reactivity control and power distribution. The limiting conditions for operation are verified by the performance of surveillance requirements. Surveillance requirements specify the tests and functional requirements that equipment must satisfy in order to meet the limiting conditions for operation.

The significance of the technical specifications is further realized through evaluation of the bases for the specifications. Each technical specification basis provides information as to why some limit or requirement has been established and how it contributes to plant safety.

### 3.1.2 Technical Specification Requirements

Each utility applying for a license to operate a nuclear reactor for commercial purposes must submit proposed technical specifications with its application to the Nuclear Regulatory Commission. This is a requirement of the Code of Federal Regulations, Title 10, Chapter I, Part 50, paragraph 36 (10 CFR 50.36). Each license authorizing operation of a production facility will then include technical specifications. Technical specifications are derived from analyses and evaluations included in the Final Safety Analysis Report (FSAR).

10 CFR 50.36 requires that technical specifications include the following categories:

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

Safety limits are limits upon process variables necessary to protect the integrity of certain physical barriers which guard against the uncontrolled release of radioactivity. The physical barriers protected by safety limits are the fuel cladding and the reactor coolant system (RCS).

Limiting safety system settings are settings for automatic protective devices related to variables having significant safety functions. The protective action from a limiting safety system setting corrects the abnormal situation before the safety limits are exceeded.

Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation. Each LCO is accompanied by applicability and action statements. The applicability statement identifies plant conditions during which the limiting condition for operation applies. The action statements provide requirements that are invoked when the limiting condition for operation is not met. The lowest allowed performance level of a system is met when either an action statement of the LCO or the LCO itself is satisfied.

Surveillance requirements are requirements relating to tests, calibrations, and inspections which ensure that the LCOs are met.

Design features included in technical specifications are those features of the facility which, if altered or modified, would have a significant effect on safety. Such features include construction materials and geometric arrangements.

Administrative controls are provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to ensure safe operation.

The Code of Federal Regulations (10 CFR 50.36a) separately requires technical specifications for radioactive effluents. Radioactive effluent technical specifications require compliance with the dose limits of 10 CFR 20, procedures and equipment for the control of radioactive effluents, and reporting requirements for radioactive releases. These requirements are addressed in the administrative controls section of the technical specifications, which is discussed in Section 3.3 of this manual.

Other sections included in technical specifications are (1) definitions and (2) the bases for the safety limits, limiting safety system settings, LCOs, and surveillance requirements. The definitions section provides meanings for expressions used throughout the technical specifications. Bases are summary statements of the bases or reasons for specifications. Bases are required to be submitted with technical specifications for license application, but are not required as a part of technical specifications. Nevertheless, they are routinely included in technical specifications.

### 3.1.3 Technical Specification Formats

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.

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. Attachment A illustrates a custom LCO and accompanying surveillance requirements for the safety injection system.

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 Westinghouse plants, the standard format and content of technical specifications was provided in NUREG-0452, “Standard Technical Specifications for Westinghouse Pressurized Water Reactors” (revised several times). 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. Attachment B illustrates a standard LCO and accompanying surveillance requirements for the emergency core cooling systems (ECCSs).

Over the last few decades there has been a trend toward including in technical specifications not only those requirements derived from the analyses and evaluations in the FSAR, but also essentially all other NRC requirements governing reactor operation. This extensive use of technical specifications has been due in part to a lack of well-defined criteria for what should be included in technical specifications. This practice has contributed to the volume of technical specifications, a large increase in the number of technical specification amendment applications, and a potentially adverse impact on safety.

To address these issues, in the 1980s the nuclear industry and the NRC began studying whether the existing technical specification requirements needed improvement. This effort culminated in 1993 with the issuance of revised criteria for the contents of technical specifications, published in the “Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors.” The final policy statement was incorporated into 10 CFR 50.36 in 1995 and reads as follows:

> (ii) 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:
>
> > (A) *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.
> >
> > (B) *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.
> >
> > (C) *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.
> >
> > (D) *Criterion 4*. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Based on the criteria of the final policy statement (and the preceding interim criteria), improved standard technical specifications have been developed for each NSSS design. These improved standard specifications are the result of extensive public technical meetings and discussions between the NRC staff and various nuclear power plant licensees, NSSS owners groups, NSSS vendors, and the Nuclear Energy Institute. For Westinghouse pants, the improved standard technical specifications are provided in NUREG-1431, “Standard Technical Specifications, Westinghouse Plants,” originally issued in 1992 and first revised in 1995. Attachment C illustrates an improved standard LCO and accompanying surveillance requirements for the ECCSs.

Licensees are encouraged to upgrade their technical specifications consistent with the criteria of the final policy statement and conforming to the improved standard technical specifications. The NRC continues to place the highest priority on requests for complete conversions to the improved standard technical specifications. Several licensees have already converted their technical specifications, and it is expected that ultimately most specifications will conform to the improved standard format. Technical specifications conforming to the improved standard technical specifications of NUREG-1431 have been developed for TTC Unit 2 (the Westinghouse simulator) and are used to illustrate technical specification requirements and usage in this manual.

### 3.1.4 Definitions

To ensure a uniform interpretation of technical specifications, selected terms are defined in the definitions section. The definitions comprise a subsection of the use and application section of the specifications. Defined terms used throughout technical specifications are identified by upper-case type (a practice also observed in the technical specification sections of this manual). Selected definitions are discussed in the following paragraphs.

#### 3.1.4.1 Instrumentation

The proper measurement of process variables such as pressurizer pressure, RCS average temperature ($T_{avg}$), RCS differential temperature ($\Delta T$), and nuclear power is verified by three methods:

- CHANNEL CHECK: the qualitative assessment of channel behavior during operation by observation.
- CHANNEL OPERATIONAL TEST: the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of the required alarm, interlock, display, and trip functions.
- CHANNEL CALIBRATION: the adjustment, as necessary, of the channel output such that it responds within the required range and accuracy to known values of input.

Control room operators perform CHANNEL CHECKS during routine observation of control board indications. For instance, the indications of all four RCS $T_{avg}$ meters are compared to each other to ensure agreement among them. If a deviation exists, instrumentation and control technicians then perform a CHANNEL OPERATIONAL TEST on the instrument showing the deviation. CHANNEL OPERATIONAL TESTS are performed on all channels which measure process variables at routine intervals and when deviations are identified. Adjustments of trip, interlock, and alarm setpoints are made to ensure that the setpoints are within the required range of accuracy. A CHANNEL CALIBRATION is performed to ensure that a channel responds properly to a known input. During a channel calibration, the channel's sensor and bistables are adjusted to provide accurate and proper responses.

#### 3.1.4.2 RCS Leakage

Because of the potential radiological and equipment problems associated with RCS LEAKAGE, it is important to understand the types of LEAKAGE. These include:

- Identified LEAKAGE,

- Unidentified LEAKAGE, and
- Pressure boundary LEAKAGE.

Identified LEAKAGE is LEAKAGE:

Such as that from pump seals or valve packing, that is captured and conducted to collection systems or a sump or collecting tank;

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

From the RCS through a steam generator to the secondary system.

Pressure boundary LEAKAGE is LEAKAGE (except steam generator tube LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

Unidentified LEAKAGE is leakage which is not identified LEAKAGE. When LEAKAGE from the RCS is first suspected or discovered, it is normally categorized as unidentified LEAKAGE.

Once the source of LEAKAGE is determined, it can be recategorized as either identified or pressure boundary LEAKAGE.

### 3.1.4.3 Operability

Many limiting conditions for operation for specific equipment require systems or components to be OPERABLE. The designation OPERABLE/OPERABILITY stipulates that a system, subsystem, train, component, or device is capable of performing its specified safety functions, and that all necessary controls, power, and auxiliary equipment required for the system, subsystem, train, component, or device to perform its specified safety functions are also capable of performing their related support functions. For instance, a centrifugal charging pump that starts and delivers flow to the RCS could not be considered OPERABLE if its lubricating oil or cooling water support system is out of service.

### 3.1.4.4 Operational Modes

Operation of the plant is divided into six operational MODES. An operational MODE corresponds to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning, as specified in technical specification Table 1.1-1, with fuel in the reactor vessel.

Operational MODES are used to identify plant conditions during which certain limiting conditions for operation apply.

### 3.1.4.5 Shutdown Margin

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM, and

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.

### 3.1.5 Use and Application

Section 1.0, Use and Application, contains definitions (as in the examples above) and also gives several examples of how to use technical specifications.

Section 1.2 explains the use of the logical connectors AND and OR. Levels of logic are identified by the placement (or nesting) of the logical connectors.

Section 1.3 establishes the completion time convention and gives several examples of completion time application. The completion time extension associated with subsequent inoperability is defined here.

Section 1.4 defines the proper use and application of the frequency requirements and gives several examples.

It is not the intention of this manual to repeat these sections of technical specifications. The student should read section 1.0 and work through the examples.

### 3.1.6 Safety Limits and Limiting Safety System Settings

#### 3.1.6.1 Safety Limits

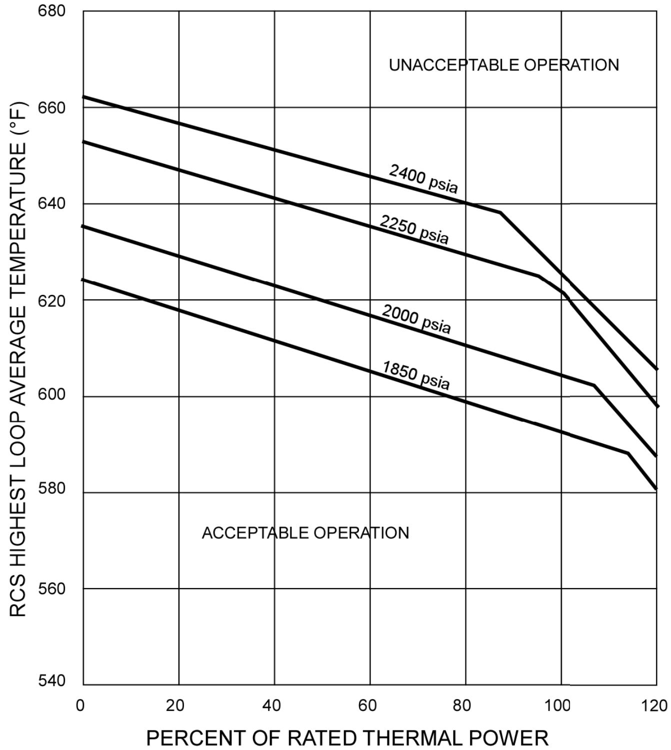
Safety limits are established to prevent the uncontrolled release of radioactivity by protecting the integrity of fission product barriers during normal operation and anticipated operational occurrences.

The safety limits on the reactor core address the first barrier to the release of radioactive material, the fuel cladding. The possibility of fuel cladding damage is prevented by observing operating limits that preclude violation of the following fuel design criteria:

There must be at least 95% probability at a 95% confidence level (the 95/95 departure from nucleate boiling [DNB] criterion) that the hot fuel rod in the core does not experience DNB; and the hot fuel pellet in the core must not experience centerline fuel melting.

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core safety limits.

The safety-limit curves in Figure COLR-1 show the loci of points of thermal power, pressure, and average coolant temperature for which either:




- The average enthalpy at the core exit is equal to the enthalpy of saturated liquid (left-hand segments),
- The minimum departure from nucleate boiling ratio (DNBR) is not less than the safety-analysis limit (right-hand segments), or
- The local core-exit quality is within the limits defined in the DNBR correlation (center segments of some curves).

Operation of the plant with a $T_H$ less than saturation temperature results in a core ΔT ($T_H$ - $T_c$) proportional to reactor power. Consequently, the measured ΔT can be used as an input to the reactor protection system for the prevention of overpower conditions. If the average core-exit temperature reaches saturation, the core ΔT is no longer proportional to power. With the hot leg at saturation, an increase in power increases steam formation (i.e., increases the steam fraction at the core exit) without an increase in ΔT. To prevent this condition, hot-leg saturation is a limiting factor for the safety-limit curves.

The DNBR limit is imposed because of the rapid and large increase in cladding wall temperature that accompanies the departure from nucleate boiling. (Refer to Section 3.4 of this manual for a discussion of DNB and DNBR.)

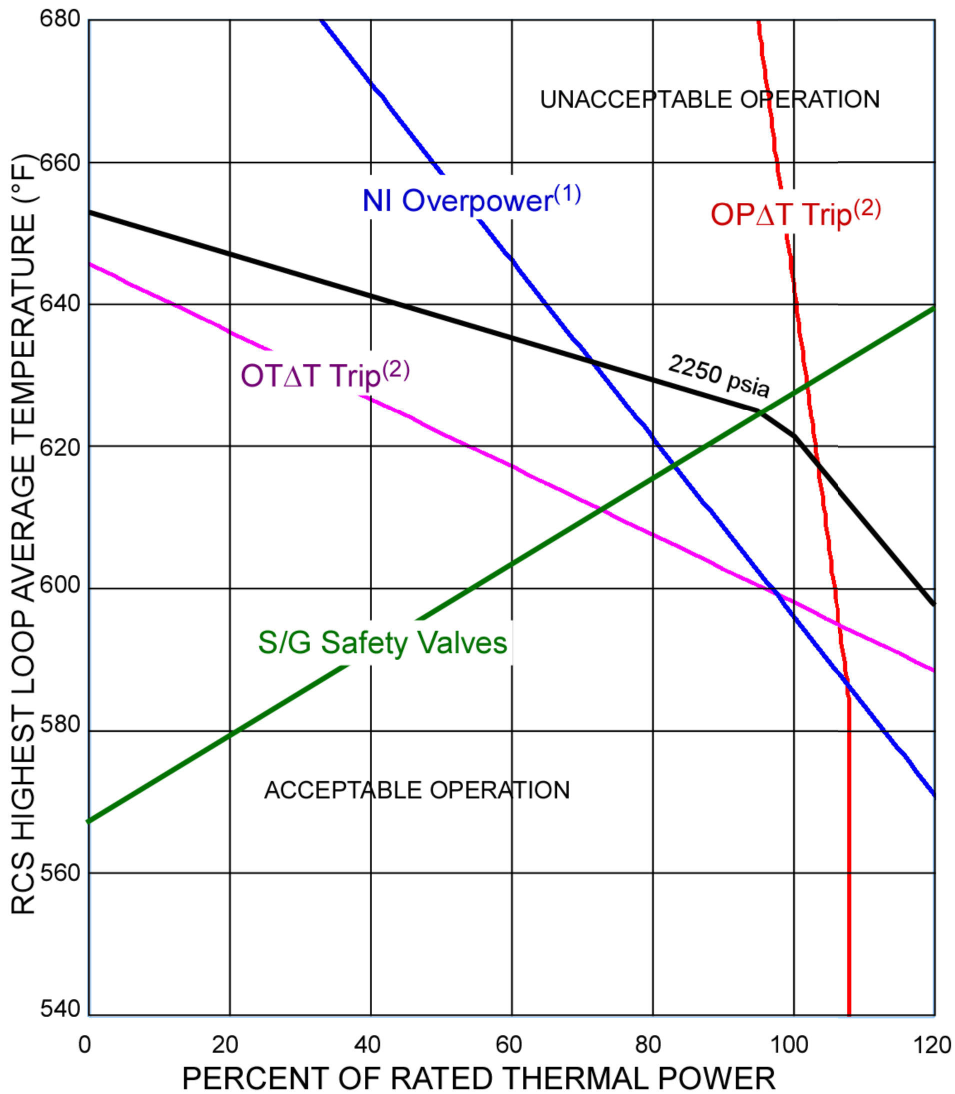
The safety limit on RCS pressure protects the integrity of the RCS. The limiting pressure for the RCS is the maximum pressure allowed in the reactor vessel by Section III of the ASME Code for Nuclear Power Plant Components. This pressure is 2735 psig, which corresponds to 110% of design pressure. By maintaining the integrity of the reactor vessel and the reactor coolant piping, valves, and fittings, the release of radionuclides to the containment atmosphere is prevented.

If any safety limits is violated during power operation, the licensee must restore compliance in accordance with SL 2.2. In addition to this, the licensee must comply with 10 CFR 50.36(c)(1) which includes the requirement that “Operation must not be resumed until authorized by the Commission.”. The licensee must also notify the NRC Operations Center within one hour, in accordance with 10 CFR 50.72.

### 3.1.6.2 Limiting Safety System Settings

Limiting safety system settings are protective device setpoints selected to prevent the reactor core and RCS from exceeding their safety limits during normal operation and anticipated operational occurrences (Condition I and II events). In standard technical specifications (conforming to NUREG-0452), the limiting safety system settings are listed in a separate section. In improved standard technical specifications (conforming to NUREG-1431), the limiting safety system settings are included as the reactor trip system (RTS) setpoints in the RTS instrumentation LCO. This LCO is discussed in Section 3.2 of this manual.

Figure 3.1-1 illustrates protective features designed to prevent exceeding the DNB safety limit.




(1) NI trip setpoint adjusted for changes in neutron shielding due to changes in vessel downcomer temperature
(2) Setpoint does not include dynamic compensation

Figure 3.1-1 Reactor Core Safety Limits vs. Boundary of Protection

### 3.1.7 Limiting Conditions For Operation

Limiting conditions for operation provide the lowest functional performance levels required of equipment for safe operation.

Attachment A is an example of a custom technical specification LCO. Note that the statement of the LCO in the left-hand column specifies the extent of system OPERABILITY and the operational MODES in which the LCO is applicable. Note also that some of the left-hand column paragraphs provide additional requirements and time limits for action in the event that the LCO is not completely satisfied. The surveillance requirements applicable to each paragraph of the LCO are provided in the right-hand column. The bases for this LCO (not shown in Attachment A) and other ECCS LCOs are provided at the end of the ECCS section of the technical specifications.

Attachment B is an example of a standard technical specification LCO. The statement of the LCO, the applicability requirements, and the actions to be taken when the LCO is not met are provided in a more straightforward manner. This format for LCO presentation is consistently maintained throughout the technical specifications. The LCO and its associated applicability and action statements are immediately followed by the surveillances required to ensure that the LCO is satisfied (only the first page of surveillance requirements is shown in Attachment B). The bases for this LCO (also shown in Attachment B) and other ECCS LCOs are provided in the ECCS bases section of the specifications.

The LCO section of standard technical specifications (conforming to the NUREG-0452 format) is typically divided into the following subsections:

- Applicability,
- Reactivity control systems,
- Power distribution limits,
- Instrumentation,
- Reactor coolant system,
- Emergency core cooling systems,
- Containment systems,
- Plant systems,
- Electrical power systems,
- Refueling operations,
- Special test exceptions,
- Radioactive effluents, and
- Radiological environmental monitoring.

The bases section of the specifications immediately follows the last LCO. The bases section is divided into subsections consistent with those listed above for the LCOs.

Attachment C is an example of an improved standard technical specification LCO. The LCO and applicability statements are similar to those of standard technical specification LCOs, but the actions required when the LCO is not met and the completion times for those actions are presented in tabular form. The surveillance requirements, which immediately follow the LCO, are also presented in tabular form.

The use and application section of the technical specifications provides guidance for the interpretation of the required actions and completion times included in the LCO action tables, and also for the frequencies with which required surveillances must be performed.

The LCO section of improved standard technical specifications (conforming to the NUREG-1431 format) is typically divided into the following subsections:

- Applicability,
- Reactivity control systems,
- Power distribution limits,
- Instrumentation,
- Reactor coolant system,
- Emergency core cooling systems,
- Containment systems,
- Plant systems,
- Electrical power systems, and
- Refueling operations.

These LCO groupings are similar to those of standard technical specifications. In improved standard technical specifications, special test exceptions are incorporated into the reactivity control systems section, and requirements for radioactive effluents and radiological environmental monitoring are incorporated into programs required by the administrative controls section.

The bases section of improved standard technical specifications is divided into subsections consistent with the list above and is provided in a separate specification volume. The basis for a particular LCO provides information in the following areas:

- Background information on the subject system, component, or parameter;
- How the LCO relates to applicable safety analyses in the FSAR;
- The contribution to unit safety provided by compliance with the LCO;
- The conditions in which the LCO applies;
- Reasons for the required actions to be taken when the LCO is not met;
- Reasons for the surveillance requirements which verify compliance with the LCO; and
- Referenced documents.

This information is generally much more extensive and far more indicative of the LCO's relationship to safety analyses than that provided by the superseded standard technical specifications.

Detailed discussions of the LCOs in each LCO subsection of the technical specifications are provided in Sections 3.2, 3.3, and 3.4 of this manual.

### 3.1.8 Technical Requirements Manual

Many LCOs which had been included in standard technical specifications (conforming to NUREG-0452) do not meet the updated criteria for inclusion in technical specifications and are thus not included in improved standard technical specifications (conforming to NUREG-1431). Many of these LCOs and their associated action and surveillance requirements have been relocated to the Technical Requirements Manual. These requirements are implemented in the same fashion as technical specifications, but they are treated as plant procedures. Violations of technical requirement action or surveillance requirements are not reportable as conditions prohibited by technical specifications per 10 CFR 50.72 or 10 CFR 50.73. Also, power reductions or plant shutdowns required to comply with technical requirement action statements are not reportable per 10 CFR 50.72 or 10 CFR 50.73. Violations of technical requirement action or surveillance requirements are treated as plant procedure violations by licensees and may be cited as such by NRC inspectors.

### 3.1.9 Exercises

Exercise 1

The unit is operating at 95% power. On May 15 at 1:00 p.m., accumulator A becomes inoperable because its boron concentration is not within limits. On May 17 at 2:00 p.m., accumulator D becomes inoperable for the same reason. On May 17 at 4:00 p.m., the boron concentration of accumulator A is restored to within limits (i.e., the operability of accumulator A is restored). See LCO 3.5.1.

1. At the time the first accumulator becomes inoperable, what condition is entered?
2. At the time the second accumulator becomes inoperable, what condition(s) apply?
3. When the first inoperable accumulator is restored to operable status, what condition(s) apply?
4. How long can accumulator D remain inoperable before a condition requiring a unit shutdown is entered?

Exercise 2

The unit is in Mode 1. The following sequence of events occurs (see LCO 3.8.1):

| | |
|---|---|
| June 1, 8:00 a.m. | Diesel generator B becomes inoperable. |
| June 3, 8:00 a.m. | The offsite circuit which supplies power to ESF bus A becomes inoperable. The diesel generator remains inoperable. |
| June 3, 4:00 p.m. | Diesel generator B is restored to operable status. The offsite circuit remains inoperable. |
| June 5, 8:00 a.m. | Diesel generator A becomes inoperable. The offsite circuit remains inoperable. Assume the bus remains energized through the unit auxiliary transformer. |
| June 5, 2:00 p.m. | The inoperable offsite circuit is restored to operable status. Diesel generator A remains inoperable. |

1. State the conditions which apply at each interval.
2. How long can diesel generator A remain inoperable before a condition requiring a unit shutdown is entered?

Exercise 3

On May 8 at 6:00 p.m., with the unit at 90% power, a power range neutron flux channel becomes inoperable, necessitating the performance of Surveillance Requirement 3.2.4.2 for verification of the quadrant power tilt ratio (see LCO 3.2.4). The first two verifications are made at the following times: May 9 at 8:00 a. m., and May 9 at 10:00 p.m. Does either of these violate the specified surveillance Frequency?

## TECHNICAL SPECIFICATIONS UNIT 1 - EXERCISE 1 SOLUTION

1. At the time the first accumulator becomes inoperable, what condition is entered?

Condition A of LCO 3.5.1 (one accumulator inoperable due to boron concentration not within limits) is entered.

2. At the time the second accumulator becomes inoperable, what condition(s) apply?

Condition D of LCO 3.5.1 is entered, because two accumulators are now inoperable. Also, the Completion Time for Condition A continues to be tracked.

3. When the first inoperable accumulator is restored to operable status, what condition(s) apply?

When the first inoperable accumulator is restored to operable status, Condition D is exited; operation continues in accordance with Condition A. The unit has been in Condition A for 51 hours.

4. How long can accumulator D remain inoperable before a condition requiring a unit shutdown is entered?

In accordance with the rules for Completion Times and Completion Times example 1.3-2 in the Technical Specifications, the Completion Time for Condition A may be extended if the accumulator restored to operable status is the first inoperable accumulator. A 24-hour extension to the stated 72 hours is allowed, provided that accumulator D does not remain inoperable for greater than 72 hours. Extending the Completion Time by 24 hours allows Condition A to remain in effect until 1:00 p.m. on May 19, at which time accumulator D will have been inoperable for 47 hours. If the boron concentration of accumulator D is not restored by then, Condition C, a condition requiring a unit shutdown, will be entered.

# TECHNICAL SPECIFICATIONS UNIT 1 - EXERCISE 2 SOLUTION

1. State the conditions which apply at each interval.

June 1, 8:00 a.m. Condition B is entered. Completion Time for restoration of operable status (Required Action B.4): 72 hours. (Other required actions apply.)

June 3, 8:00 a.m. Conditions A & D are entered. Completion Time for restoration of offsite circuit operability (Required Action A.3): 72 hours. (Other required actions for Condition A apply.) Completion Time for restoration of either diesel generator or offsite circuit (Required Action D.1 or D.2): 12 hours. Condition B still applies; the unit has been in Condition B for 48 hours.

June 3, 4:00 p.m. Conditions B & D are exited (each within the specified Completion Time). Condition A still applies; the unit has been in Condition A for 8 hours.

June 5, 8:00 a.m. Conditions B & D are reentered, with Completion Times of 72 hours and 12 hours, respectively. Condition A still applies; the unit has been in Condition A for 48 hours. Since the bus still has an AC power source (the unit auxiliary transformer), it is not necessary to declare the bus inoperable (i.e. apply actions of LCO 3.8.9). A bus is operable if it is energized, even if it has neither TS required source.

June 5, 2:00 p.m. Conditions A & D are exited (each within the specified Completion Time). Condition B still applies; the unit has been in Condition B for 6 hours.

2. How long can diesel generator A remain inoperable before a condition requiring a unit shutdown is entered?

Diesel generator A can remain inoperable until 8:00 a.m. on June 8, when its 72 hour completion time expires. At that time, condition G is entered.

**TECHNICAL SPECIFICATIONS UNIT 1 - EXERCISE 3 SOLUTION**

Does either of these violate the specified surveillance Frequency?

No. In accordance with Frequency example 1.4-3 in the Technical Specifications, the 25% extension can be applied to the first performance, and to subsequent intervals. This allows a 15 hour interval. The Frequency is satisfied for both performances.

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# ATTACHMENT A - CUSTOM TECHNICAL SPECIFICATION

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.8.2 Safety injection pump system | 4.8.2 Safety injection pump system |
| A. The two safety injection systems shall be operable whenever the reactor is going from hot shutdown to hot standby. | A. Surveillance and pump testing of the safety injection system shall be performed as follows:<br>1. The safety injection pumps shall be started manually from the control room each month. Performance will be acceptable if the pump starts upon actuation, operates for at least 10 minutes on recirculation flow, and the discharge pressure and recirculation flow are within ± 10% of a point on the pump head curve.<br>2. The annunciators associated with the normally open valve (MOV-SI8806) in the suction of the safety injection pumps shall be checked quarterly.<br>3. The normally open valve (MOV-SI8806) in the suction line of the safety injection pumps shall be stroked manually from the control room to check the position indicators and annunciators every refueling outage. |
| B. The two safety injection systems shall be operable whenever the reactor is in hot standby or operating except as specified in 3.8.2.C. | B. Not Applicable |
| C. From and after the date that one of the two safety injection pumps is made or found to be inoperable for any reason, reactor operation including recovery from an inadvertent trip is permissible only during the succeeding 7 days provided that during those 7 days the remaining safety injection pump system and both centrifugal charging pump systems and both residual heat removal pump systems are operable. | C. When it is determined that one of the two safety injection pump systems is inoperable the remaining safety injection pump system, both centrifugal charging pump system, and both residual heat removal pump systems, including the associated standby AC and DC power supplies (see sections 4.15.1.B.2 and 4.15.1.B.1) shall be demonstrated to be operable immediately and daily thereafter. |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.8.2 D. If these conditions cannot be met the reactor shall be brought to the hot shutdown condition within four hours. After a maximum of 48 hours in the hot shutdown condition, if the system is not operable the reactor shall be brought to the cold shutdown condition within 12 hours. | 4.8.2 D. Not Applicable |

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE safety injection pump,

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the following valves are in the indicated position with power to the operators removed:

| Valve Number | Valve Function | Valve Position |
|---|---|---|
| MO 8806 | RWST Isolation | open* |
| MO 8812 | RHR Suction | open* |
| MO 8835 | SIS Cold Leg Injection | open* |
| MO 8802-A | SIS Hot Leg Injection | closed |
| MO 8802-B | SIS Hot Leg Injection | closed |
| MO 8703 | RHR Hot Leg Discharge | closed |
| MO 8809-A | RHR Cold Leg Discharge | open* |
| MO 8809-B | RHR Cold Leg Discharge | open* |
| MO 8811-A | Recirc. Sump, RHR Suction | closed* |
| MO 8811-B | Recirc. Sump, RHR Suction | closed* |
| MO 8813 | SI Pump Mini-flow Isolation | open* |
| MO 8814 | SI Pump Mini-flow Isolation | open* |

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

   1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

   2. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

*Power to be restored and valves operated from within control room for switch from injection to recirculation mode following LOCA.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.2 and 3/4.5.3.1 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without a single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements, which are provided to ensure the OPERABILITY of each component, ensure that, at a minimum, the assumption used in the safety analysis are met and that subsystem OPERABILITY is maintained. The safety analyses make assumptions with respect to: (1) both the maximum and minimum total system resistance and (2) both the maximum and minimum branch injection line resistance. These resistances, in conjunction with the ranges of potential pump performance, are used to calculate the maximum and minimum ECCS flow assumed in the safety analysis.

The maximum and minimum flow Surveillance Requirements in conjunction with the maximum and minimum pump performance curves ensure that the assumptions of total system resistance and the distribution of that system resistance among the various paths are met.

The maximum total pump flow Surveillance Requirements ensure that the pump runout limits of 560 gpm for the centrifugal charging pumps and 675 gpm for the safety injection pumps are not exceeded.

The Surveillance Requirements for the maximum difference between the maximum and minimum individual injection line flows ensure that the minimum individual injection line resistance assumed for the spilling line following a LOCA is met.

The safety analyses are performed assuming the miniflow recirculation lines for the ECCS subsystems associated with the centrifugal charging and safety injection pumps are open. The flow balancing test is, therefore, performed with these miniflow recirculation lines open.

The surveillance flow and differential pressure requirements are the Safety Analysis Limits and do not include instrument uncertainties.

# ATTACHMENT C - IMPROVED STANDARD TECHNICAL SPECIFICATION

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

-------------------------------------------NOTES-------------------------------------------

1. In MODE 3, both safety injection (SI) pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1.
2. In MODE 3, ECCS pumps may be made incapable of injecting to support transition into or from the Applicability of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for up to 4 hours or until the temperature of all RCS cold legs exceeds 315°.

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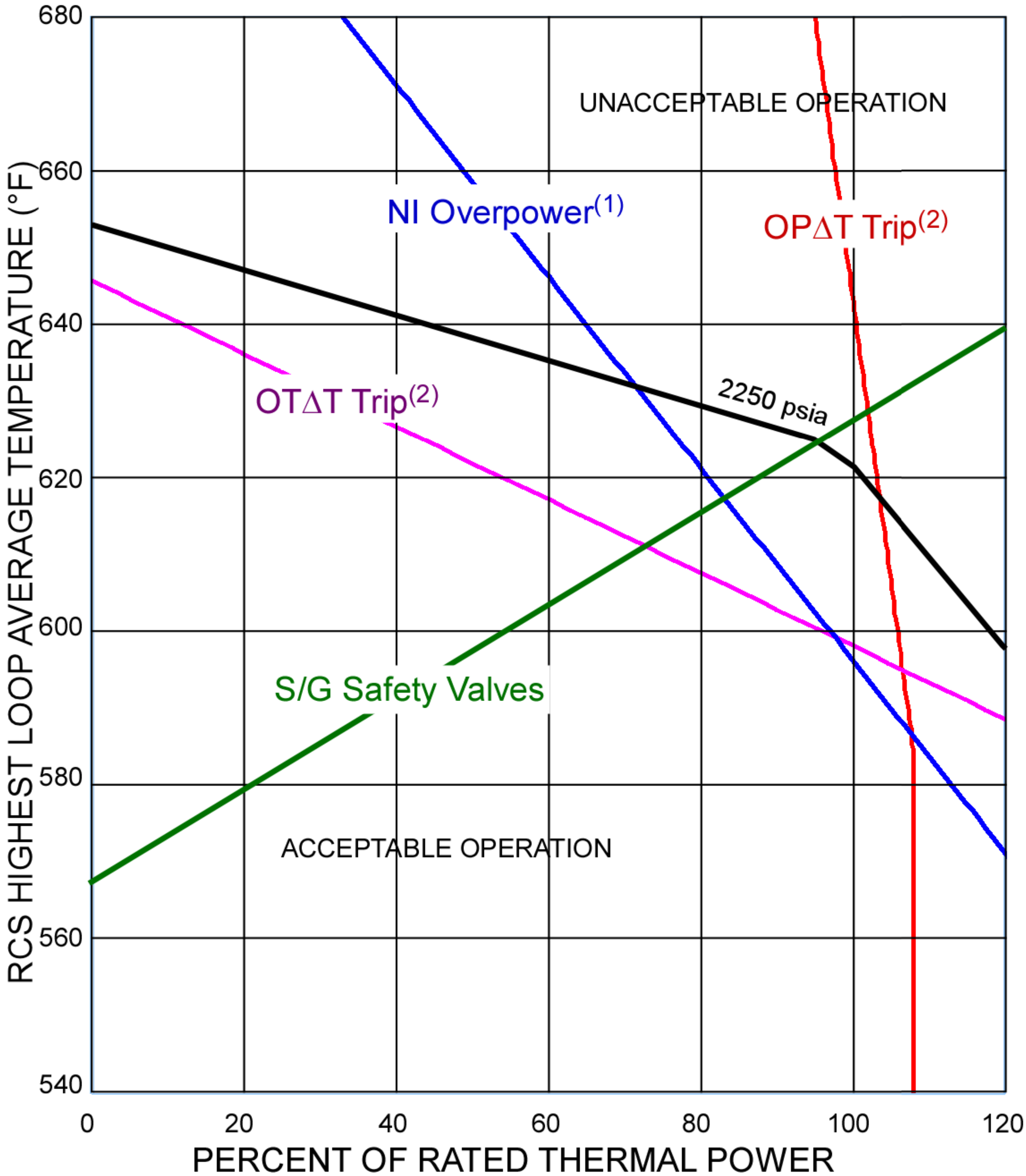
APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more trains inoperable. | A.1 Restore train(s) to OPERABLE status. | 72 hours |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br><br><u>AND</u><br><br>B.2 Be in MODE 4. | 6 hours<br><br><br><br>12 hours |
| C. Less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available. | C.1 Enter LCO 3.0.3. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.5.2.1 | Verify the following valves are in the listed position with power to the valve operator removed.<br><br><u>Number</u> / <u>Position</u> / <u>Function</u><br>MO 8806 / Open / SIS RWST Iso.<br>MO 8812 / Open / RHR RWST Iso<br>MO 8835 / Open / SIS Cold Leg Inj.<br>MO 8802-A, B / Closed / SIS Hot Leg Inj.<br>MO 8703 / Closed / RHR Hot Leg Disch.<br>MO 8809-A, B / Open / RHR Cold Leg Disch.<br>MO 8811-A, B / Closed / RHR Recirc. Sump Iso.<br>MO 8813, 8814 / Open / SI Pump Mini-flow Iso. | 12 hours |
| SR 3.5.2.2 | Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position. | 31 days |
| SR 3.5.2.3 | Verify ECCS piping is full of water. | 31 days |
| SR 3.5.2.4 | Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head. | In accordance with the Inservice Testing Program |
| SR 3.5.2.5 | Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. | 18 months |
| SR 3.5.2.6 | Verify each ECCS pump starts automatically on an actual or simulated actuation signal. | 18 months |
| SR 3.5.2.7 | Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion. | 18 months |




(1) NI trip setpoint adjusted for changes in neutron shielding due to changes in vessel downcomer temperature

(2) Setpoint does not include dynamic compensation

**Figure 3.1-1 Reactor Core Safety Limits vs. Boundary of Protection**

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