**Florida Power**
A Progress Energy Company

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: ITS 5.6.2.17

May 31, 2002
3F0502-06

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Crystal River Unit 3 – Technical Specifications Bases Control Program

Dear Sir:

Florida Power Corporation (FPC) hereby submits the changes that were made to the Crystal River Unit 3 (CR-3) Improved Technical Specifications (ITS) Bases as required by ITS 5.6.2.17. The attachments provide revisions to the CR-3 ITS Bases that will update NRC copies of the ITS.

Attachment A provides the instructions for updating the CR-3 ITS. Attachment B provides the ITS and Bases List of Effective Pages. Attachment C provides the replacement pages for the CR-3 ITS Bases.

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing and Regulatory Programs at (352) 563-4883.

Sincerely,

SL Bernhoft

S. L. Bernhoft
Manager Regulatory Affairs

SLB/ff

Attachments:

A. Instructions for Updating the Crystal River Unit 3 ITS and Bases
B. ITS and Bases List of Effective Pages
C. Replacement ITS Bases Pages

xc: NRR Project Manager (w/o Attachment C)
Regional Administrator, Region II (w/o Attachment C)
Senior Resident Inspector (w/o Attachment C)

Crystal River Nuclear Plant
15760 W. Power Line Street
Crystal River, FL 34428

A001

**FLORIDA POWER CORPORATION**

**CRYSTAL RIVER UNIT 3**

**DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72**

**ATTACHMENT A**

# INSTRUCTIONS FOR UPDATING THE CRYSTAL RIVER UNIT 3 IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

INSTRUCTIONS FOR UPDATING
THE CRYSTAL RIVER UNIT 3
IMPROVED TECHNICAL SPECIFICATIONS

5/31/02

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THE CRYSTAL RIVER UNIT 3
IMPROVED TECHNICAL SPECIFICATIONS

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**FLORIDA POWER CORPORATION**

**CRYSTAL RIVER UNIT 3**

**DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72**

# ATTACHMENT B

# ITS AND BASES LIST OF EFFECTIVE PAGES

# IMPROVED TECHNICAL SPECIFICATIONS

## List of Effective Pages
## (Through Amendment 202)

***Amendment Nos. 159, 164, 166, 171, 173, 181, 189 and 190 amended the CR-3 Operating License, only, and did not effect changes to the ITS LCOs or Bases.***

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**FLORIDA POWER CORPORATION**

**CRYSTAL RIVER UNIT 3**

**DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72**

**ATTACHMENT C**

**REPLACEMENT ITS BASES PAGES**

# B 2.0 SAFETY LIMITS (SLs)

## B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

Crystal River Unit 3 FSAR Section 1.4 (Ref. 1) Criterion 6 requires that acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences (AOOs). The reactor core SLs are established to preclude violation of the following fuel design criteria:

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

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

DNB cannot be measured directly during power operation. However, THERMAL POWER, reactor coolant pressure, temperature, flow and power peaking can be correlated to the critical heat flux (CHF), which is the heat flux at which DNB occurs. CHF correlations have been developed experimentally to predict the departure from nucleate boiling ratio (DNBR), defined as the ratio of the heat flux required to cause DNB at a particular core location to the local heat flux. The DNBR is an indication of the margin to DNB for core design purposes and safety analysis evaluations.

The restrictions of this SL provide a high degree of protection against overheating of the fuel and cladding that would result in possible cladding perforation. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

The 95/95 DNB criterion is preserved by ensuring that the DNBR remains greater than the DNBR design limit based upon the applicable CHF correlation for the core design. In the development of the applicable DNBR design limit, uncertainties in the core state variables, power peaking

(continued)

BASES

BACKGROUND (continued)

factors, manufacturing related parameters, and the CHF correlation may be statistically combined to determine a statistical DNBR design limit. Additional retained thermal margin may also be applied to the statistical DNBR design limit to yield a higher thermal design limit for use in establishing DNB-based core safety and operating limits. In all cases, application of statistical DNBR design methods preserves the 95/95 DNB criterion.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The melting point of uranium dioxide varies slightly with burnup. As uranium is depleted and fission products produced, the net effect is a decrease in the melting point. However, depletion of the uranium also reduces the power produced in the fuel such that the closest the plant comes to the centerline melt SL is at the beginning of the fuel cycle. The formula presented is on a per fuel pin basis.

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The RPS setpoints (Ref. 2), in combination with the DNB operating limits LCO (LCO 3.4.1), are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

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

a. RCS High Pressure trip;

b. RCS Low Pressure trip;

c. Nuclear Overpower trip;

d. RCS Variable Low Pressure trip;

e. Reactor Coolant Pump to Power trip; and

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

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

Safety Limits that preclude fuel cladding failure are required to be included in the Technical Specifications pursuant to 10 CFR 50.36 (Ref. 5).

SAFETY LIMITS

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature stays below the melting point, or the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or the exit quality is within the limits defined by the DNBR correlation. In addition, SL 2.1.1.3 addresses the pressure/temperature operating region that keeps the reactor from reaching an SL when operating up to design power.

(continued)

BASES (continued)

SAFETY LIMITS

Examination of the limit curve in Figure 2.1.1-1 reveals that the temperatures corresponding to the pressures vary between 20 and 30°F below the saturation temperature of the coolant at that pressure, thus ensuring an even greater margin to DNB.

The fuel centerline melt and DNBR SLs are not directly monitorable by installed plant instrumentation. Instead, the SLs are preserved by monitoring the process variable AXIAL POWER IMBALANCE to ensure that the core operates within the fuel design criteria. With AXIAL POWER IMBALANCE within the protective limits, fuel centerline temperature and DNBR are also within limits. AXIAL POWER IMBALANCE protective limits are provided in the COLR.

The AXIAL POWER IMBALANCE protective limits are preserved by their corresponding RPS setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," as specified in the COLR. The trip setpoints are derived by adjusting the measurement system independent AXIAL POWER IMBALANCE protective limit given in the COLR to allow for measurement system observability (the fact there are a finite number of detectors) and instrumentation errors. The AXIAL POWER IMBALANCE protective limits are separate and distinct from the AXIAL POWER IMBALANCE operating limits defined by LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits." The AXIAL POWER IMBALANCE operating limits in LCO 3.2.3, also specified in the COLR, preserve initial conditions of the safety analyses but are not reactor core SLs.

RCS pressure, temperature and flow DNB operating limits are defined by LCO 3.4.1.

APPLICABILITY

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The automatic protection actions serve to prevent RCS heatup to reactor core SL conditions by initiating a reactor trip which forces the plant into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.

(continued)

BASES

| | |
|---|---|
| APPLICABILITY (continued) | In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER. |
| SAFETY LIMIT VIOLATIONS | The following SL violation responses are applicable to the reactor core SLs.<br><br><u>2.2.1</u><br><br>If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the requirement to go to MODE 3 places the plant in a MODE in which these SLs can not be violated.<br><br>The allowed Completion Time of 1 hour recognizes the importance of placing the plant in a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage. |
| REFERENCES | 1. FSAR, Section 1.4.<br><br>2. FSAR, Table 7-2. |

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# B 2.0 SAFETY LIMITS (SLs)

## B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

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BACKGROUND

According to FSAR Section 1.4, Criterion 9, "Reactor Coolant Pressure Boundary," is designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime (Ref. 1), the reactor coolant pressure boundary (RCPB). Criterion 33, RCPB Capability" (Ref. 1), specifies that reactivity accidents including rod ejection do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psig. During normal operation and anticipated operational occurrences (AOOs), the RCS pressure is kept from exceeding the design pressure by more than 10% in order to remain in accordance with Section III of the ASME Code (Ref. 2). Hence, the safety limit is 2750 psig. To ensure system integrity, all RCS components were hydrostatically tested at 125% of design pressure (3125 psig) prior to initial operation, according to the ASME Code requirements. Inservice operational hydrotesting in accordance with the ASME Code is also required whenever the reactor vessel head has been removed or if other pressure boundary joint alterations have occurred. Following inception of plant operation, RCS components are pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

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APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, operating in conjunction with the Reactor Protection System trip settings, ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that is most influential for establishing the required relief capacity, and hence the valve size requirements and lift settings, is a rod withdrawal from low power. Both pressurizer safety valves may be required for protection

(continued)

BASES

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APPLICABLE SAFETY ANALYSES (continued)

from this event. During the transient, no control actions are assumed except that the Reactor Protection System (RPS) trips the reactor on high flux, and nominal feedwater supply is maintained. Main Steam Safety Valves, while not specifically modelled as part of the analysis, are qualitatively assumed to function to fix secondary side pressures and temperatures (and thus, RCS cold leg temperature).

It is important to emphasize that when the operating characteristics of the safety valves were selected, it was assumed that the reactor protection system provided the first overpressure protection. The safety valves alone cannot prevent overpressure; they act in conjunction with the RPS to prevent overpressure. A single failure in the RPS will not result in a failure to trip the reactor. Failure of the RPS to trip the reactor was not assumed to be credible when the operating characteristics of the safety valves were specified. The single failure criterion is not considered to be applicable to pressurizer safety valves since the ASME code allows the use of the rated capacity of all OPERABLE spring-loaded safety valves. This allows the total relieving capacity of both valves to be credited to overpressure protection.

The overpressure protection analyses (Ref. 4) and the safety analyses (Ref. 5) are performed using conservative assumptions relative to pressure control devices. More specifically, no credit is taken for operation of the following:

a. Pressurizer power operated relief valves (PORVs);

b. Steam line turbine bypass valves;

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

d. Pressurizer spray valve.

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SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design

(continued)

BASES

SAFETY LIMITS (continued)

pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section B31.7 (Ref. 6), is also 110% of design pressure. Therefore, the SL on maximum allowable RCS pressure of 2750 psig is consistent with the design criteria and associated code requirements. Overpressurization of the RCS can result in a breach of the RCPB.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES during overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT VIOLATIONS

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

2.2.2

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 50.67 limits (Ref 7).

The allowed Completion Time of 1 hour is based on the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.3

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the RCS pressure SL in MODE 3, 4, or 5 is potentially more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. FSAR, Section 1.4.
2. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME Boiler and Pressure Vessel Code, Section XI, Articles IWA-5000 and IWB-5000.
4. BAW-10043, May 1972.
5. FSAR, Section 14.
6. ASME USAS B31.7, Code for Pressure Piping, Nuclear Power Piping, February 1968 Draft Edition.
7. 10 CFR 50.67.

## B 3.1 REACTIVITY CONTROL

## B 3.1.7 Position Indicator Channels

BASES

BACKGROUND

Criterion 12, FSAR Section 1.4 requires (Ref. 1), instrumentation be provided to monitor variables within prescribed operating ranges.

Rod position indication is needed to assess CONTROL ROD & APSR OPERABILITY and alignment. Refer to the Bases for LCO 3.1.4, 3.1.5, and LCO 3.1.6 for a more detailed discussion on why these parameters and attributes are monitored and controlled.

Two methods of CONTROL ROD and APSR position indication are provided in the CONTROL ROD Drive Control System. The two means are by absolute position indicator and relative position indicator transducers. The absolute position indicator transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the CONTROL ROD drive mechanism (CRDM) motor tube extension.

Switch contacts close when a permanent magnet mounted on the upper end of the CONTROL ROD assembly (CRA) leadscrew extension comes near. As the leadscrew and CRA move, the switches operate sequentially, adding/removing resistors from a 'bridge' arrangement and producing an analog voltage proportional to position. Other reed switches included in the same tube with the position indicator matrix provide full in and full out limit indications, and absolute position indications at 0%, 25%, 50%, 75%, and 100% travel (called zone reference indicators). The relative position indicator transducer is a potentiometer, driven by a step motor that produces a signal proportional to CONTROL ROD position, based on the electrical pulse steps that drive the CRDM.

The R4C (redundant four channel) absolute position indicator transducer has two parallel sets of voltage divider circuits made up of 36 resistors each, connected in series (channel A and B). One end of 36 reed switches is connected at a junction between each of the resistors of the two parallel circuits. The reed switches making up each circuit are offset, such that the switches for channel A are staggered with the switches for channel B. The R4C is designed such that either two or three reed switches are

(continued)

BASES

| | |
|---|---|
| BACKGROUND (continued) | closed in the vicinity of the magnet. By its design, the R4C absolute position indicator provides redundancy, with the two-three-two sequence of pickup and drop out of reed switches to enable a continuity of position signal when a single reed switch fails to close.<br><br>The API design allows for bypass of one channel (by means of a toggle switch on the amplifier card) in the event a reed switch fails.<br><br>CONTROL ROD position indicating readout devices located in the control room consist of single CRA position meters and four group average position meters on the Diamond Control Panel Section of the Main Control Board. A selector switch permits either relative or absolute position indication to be displayed on all 68 of the single rod meters. Indicator lights are provided on the single CRA meter panel to indicate when each CRA is fully withdrawn, fully inserted, enabled, or transferred, and whether a CRA position asymmetry alarm condition is present. Indicators on the console show full insertion, full withdrawal, and enabled for motion for each CONTROL ROD group. Identical instrumentation and devices exist for the APSR group. The consequence of continued operation with an inoperable absolute position indicator or relative position indicator channel is a decreased reliability in determining CONTROL ROD position. |
| APPLICABLE SAFETY ANALYSES | CONTROL ROD and APSR position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2) with CONTROL RODS or APSRs operating outside their limits undetected. Regulating rod, safety rod, and APSR positions must be known in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Safety Rod Insertion Limits"; LCO 3.2.1, "Regulating Rod Insertion Limits"; and LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "CONTROL ROD Group Alignment Limits," and LCO 3.1.6, "AXIAL POWER SHAPING |

(continued)

# B 3.3 INSTRUMENTATION

## B 3.3.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor trip, (i.e., full insertion of all CONTROL RODS) to protect against violating core fuel design limits and the Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs). By tripping the reactor, the RPS also functions in conjunction with the Engineered Safeguards (ES) Systems in mitigating accidents.

The RPS is part of a layered protection scheme designed to assure safe operation of the reactor. This defense-in-depth approach is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as providing LCOs on other reactor system parameters and equipment performance. The LSSS, defined in this Specification as the Allowable Value, in conjunction with these other LCOs, establishes the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During AOOs, (those events expected to occur one or more times during the plant's life) the RPS serves to automatically protect and maintain the following Safety Limits:

a. The departure from nucleate boiling ratio (DNBR) shall be maintained greater than the Safety Limit (SL) value of Specification 2.1.1.2;

b. Fuel centerline melt shall not occur; and

c. The RCS pressure SL of 2750 psig shall not be exceeded.

The RPS also assures offsite doses are maintained within 10 CFR 50.67 limits following accidents.

(continued)

BASES

BACKGROUND (continued)

RPS Overview

The RPS consists of four separate redundant protection channels that receive inputs of neutron flux, RCS pressure, RCS flow, RCS temperature, RC pump status, reactor building (RB) pressure, main feedwater (MFW) pump status, and main turbine status.

FSAR Figure 7-1, (Ref. 1), shows the arrangement of a typical RPS protection channel. The channel is composed of measurement channels, a manual trip channel, a reactor trip module (RTM), and CONTROL ROD drive (CRD) trip devices. LCO 3.3.1 provides requirements for the individual measurement channels. These channels encompass all equipment and electronics from the point at which the measured parameter is sensed through the bistable relay contacts in the trip string. LCO 3.3.2, "Reactor Protection System (RPS) Manual Reactor Trip," LCO 3.3.3, "Reactor Protection System (RPS)-Reactor Trip Module (RTM)," and LCO 3.3.4, "CONTROL ROD Drive (CRD) Trip Devices," discuss the remaining elements in the RPS protection channel.

An RPS instrumentation channel measures critical plant parameters (see above) and compares these to pre-determined setpoints. If the setpoint is exceeded, a channel trip signal is generated. The generation of any two trip signals in any of the four RPS channels will result in the full insertion of all CONTROL RODS. Development of the two-out-of-four logic is done in the RTM. Each RPS channel contains an RTM. The RTM receives signals from the associated measurement devices in the same channel that indicates a protection channel trip is required. The RTM transmits this signal to an internal two-out-of-four trip logic and to similar logic in the RTMs in the other three RPS channels. The two-out-of-four logic is designed such that whenever any two RPS channels sense and transmit trip signals, the RTM logic in each channel actuates to remove 120 VAC power from its associated CRD trip device(s).

(continued)

BASES

ACTIONS (continued)

A.1 and A.2

With one or more automatic actuation logic matrices inoperable, the associated component(s) should be placed in the ES configuration. This manual Action essentially fulfills the safety function of the automatic actuation logic. Placing a component in the ES configuration cannot be performed for a component that could be automatically removed from the ES configuration (example: a pump that will be tripped by undervoltage relaying during a loss of offsite power). In some circumstances, placing the component in its ES configuration would impose an undue operational restriction. In both of the above cases, Required Action A.2 allows for the component status be left as-is, and the supported system component declared inoperable. Other conditions which would potentially preclude placing of a component in its ES configuration include, but are not limited to, violation of system separation, activation of fluid systems that could lead to thermal shock, or isolation of fluid systems that are normally functioning. The 1 hour Completion Time is based on operating experience and reflects the urgency associated with the inoperability of a safety system component.

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1

SR 3.3.7.1 is the performance of a CHANNEL FUNCTIONAL TEST on a 31 day STAGGERED TEST BASIS. The CHANNEL FUNCTIONAL TEST of the Automatic Actuation Logic need only demonstrate one combination of the three two-out-of-three logic combinations that are required to be OPERABLE. A different combination is tested at each test interval, such that all three combinations will be confirmed to be OPERABLE by the time the third successive test is completed. The Frequency is based on operating experience that demonstrates the low likelihood of more than one channel failing within the same 31 day interval.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50.46.
2. FSAR, Chapter 14.

BASES

LCO (continued)

FLURs

The FLURs instrumentation associated with each ES 4160 V bus is required to be OPERABLE upon a loss of voltage. For each voltage value, the associated channel response time is based on the physical characteristics of the loss of voltage sensing relays. The loss of voltage channels respond to a complete loss of ES bus voltage, providing automatic starting and loading of the associated EDG. However, their response time is not critical to the overall ES equipment response time following an actuation, since the SLURs instrumentation will also respond to the complete loss of voltage, and will do so earlier than the loss of voltage instrumentation. Upon a complete loss of voltage from full voltage to 0.0V, the loss of voltage relays will respond in 7.8 seconds with a tolerance of 7% or ± 0.55 seconds.

APPLICABILITY

The EDG LOPS actuation Function for each EDG shall be OPERABLE in MODES 1, 2, 3, and 4 to provide protection for equipment powered from the Class 1E AC Electrical Power Distribution System in these MODES. The ability to start the EDG on a degraded or loss of voltage condition is also required for the EDG required to be OPERABLE by LCO 3.8.2, "AC Sources-Shutdown."

ACTIONS

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function. Since the required channels are specified on a per EDG basis, the Condition may be entered separately for each EDG.

A.1

A loss of one channel of loss of voltage (FLUR) Function results in a loss of redundancy for that Function, e.g., the two remaining operable channels are still capable of providing an EDG start signal assuming no additional single failures. With one channel of loss of voltage Function inoperable, the channel must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable to evaluate and take action to correct the degraded condition in an orderly manner, and is consistent with the allowed outage time for a loss of redundancy condition for other safety systems.

(continued)

BASES

ACTIONS

B.1

A loss of one or more channels of degraded voltage (SLUR) Function results in a loss of safety Function. With up to two channels of degraded voltage Function (SLUR) per EDG inoperable, the channel(s) must be tripped within 1 hour. Since there is no installed trip for the SLUR relays, a more liberal reading of the requirements is that the function of the relay must be accomplished and maintained. This involves jumpering the relay or taking other action such that the function is accomplished. The 1 hour Completion Time is reasonable to evaluate and to take action to correct the degraded condition in an orderly manner and takes into account the low probability of an event requiring this instrumentation occurring during this interval.

C.1

Condition C applies when two or more undervoltage or all three degraded voltage channels associated with a single ES 4160 V bus are operable.

With two or more FLUR channels or three SLUR channels inoperable, the logic is not capable of providing an automatic EDG LOPS signal for valid loss of voltage or degraded voltage conditions. Tripping the inoperable channels is not a viable Action for this Condition since doing so would result in an EDG start. In addition, it is unlikely that repair/restoration of multiple failed channels could be accomplished in an acceptable time frame for a condition representing a loss of the affected Function. Therefore, Required Action C.1 requires that the EDG associated with the inoperable FLUR or SLUR be declared inoperable. Depending on MODE, the Actions specified in LCO 3.8.1, "AC Sources-Operating," or LCO 3.8.2, are required to be entered immediately.

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

If the inoperable channel cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging plant systems.

E.1

Condition E is the default Condition should Required Action A.1 or B.1 not be met within the associated Completion Time.

Required Action E.1 ensures that Required Actions for affected diesel generator inoperabilities are initiated. Depending on MODE, the Actions specified in LCO 3.8.1, "AC Sources-Operating," or LCO 3.8.2, are required to be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1

A CHANNEL FUNCTIONAL TEST is performed on each required EDG LOPS channel to ensure the entire channel will perform the intended function. This test ensures functionality of each channel to output relays.

The Frequency of 31 days is considered reasonable based on the reliability of the components and on operating experience.

A Note has been added to allow performance of the SR without taking the ACTIONS for inoperable instrumentation channels although during this time period the relay instrumentation cannot initiate a diesel start. This allowance is based on the assumption that the EDG is maintained inoperable during this functional test and the appropriate actions for the inoperable EDG are entered.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.2

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The setpoints and the response to a loss of voltage and a degraded voltage test shall include a single point verification that the trip occurs within the required delay time, as shown in Reference 2.

The 18 month Frequency is based on operating experience and industry-accepted practice.

A Note has been added indicating the voltage sensing device (bus potential transformer) may be excluded from testing since these transformers are passive, inherently stable devices which cannot be calibrated. In the event of transformer failure, the corresponding degraded voltage or loss of voltage relays would trip on low voltage, actuating the associated channel (i.e., the channels fail in the safe condition). In addition, annunciation of failure of a single transformer or associated circuits would be provided via the channel monitor relay, identifying to the operator a failure within the loss of voltage or degraded voltage channels.

REFERENCES

1. FSAR, Chapter 14.

2. FSAR, Section 8.3.

BASES

ACTIONS

A.1 (continued)

With one channel inoperable, the system cannot meet the single-failure criterion and still satisfy the dual functional criteria described above. Therefore, when one vector valve logic channel is inoperable, the channel must be restored to OPERABLE status within 72 hours. This Condition is analogous to having one EFW train inoperable; wherein a 72 hour Completion Time is provided by the Required Actions of LCO 3.7.5, "EFW System." As such, the Completion Time of 72 hours is based on engineering judgment.

B.1 and B.2

If Required Action A.1 cannot be met within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.14.1

SR 3.3.14.1 is the performance of a CHANNEL FUNCTIONAL TEST every 31 days. This test demonstrates that the EFIC-EFW-vector valve logic is capable of performing its intended function. The Frequency is based on operating experience that demonstrates failure of more than one channel within the same 31 day interval is unlikely.

REFERENCES

None.

# B 3.3 INSTRUMENTATION

## B 3.3.15 Reactor Building (RB) Purge Isolation-High Radiation

BASES

BACKGROUND

The RB Purge Isolation-High Radiation Function closes the RB purge and RB mini-purge valves to isolate the RB atmosphere from the environment and minimize releases of radioactivity in the event an accident occurs.

The radiation monitoring system (RMA-1) measures the activity in a representative sample of air drawn in succession through a particulate sampler, an iodine sampler, and a gas sampler. The sensitive volume of the gas sampler is shielded with lead and monitored by a Geiger-Mueller detector. The air sample is taken from the center of the purge exhaust duct through a nozzle installed in the duct.

The monitor will alarm and initiate closure of the valves prior to exceeding the noble gas limits specified in the Offsite Dose Calculation Manual.

The closure of the purge and mini-purge valves ensures the RB remains as a barrier to fission product release. There is no bypass for this function.

APPLICABLE SAFETY ANALYSES

FSAR Chapter 14 LOCA analysis assumes RB purge and mini purge lines are isolated within 60 seconds following initiation of the event. Since the early 1980's, this isolation time has only been practically applicable to the mini-purge valves since the large purge valves are required to be sealed closed during the MODES of plant operation (1, 2, 3, and 4) in which LOCAs are postulated to occur. Even

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

for mini-purge valves, design requirements on these valves require closure times on the order of 5 seconds. Thus, the purge isolation time of the current plant design is conservative to the original safety analysis.

The signal credited for initiating purge isolation in the original safety analysis is the RB Pressure - High ESAS signal and not RB Purge Isolation - High Radiation instrumentation. As such, design basis LOCA mitigation is not a basis for including this instrumentation.

RB purge isolation on high radiation is only required to maintain 10 CFR 20 limits during normal operations. However, this is not a basis for requiring a Technical Specification. Therefore, this Specification is not required in MODES 1, 2, 3 and 4.

Closure of the purge valves on high radiation is also not credited as part of the fuel handling accident (FHA) inside containment. The activity from the ruptured fuel assembly is assumed to be instantaneously released to the atmosphere in the form of a "puff" type release. This instrumentation is retained during MODES 5 and 6 in order to allow for other RB penetrations that communicate with the RB atmosphere to be open during movement of irradiated fuel assemblies within containment. Refer to LCO 3.9.3, "Containment Penetrations" for further discussion of this allowance.

LCO

One channel of RB Purge Isolation-High Radiation instrumentation is required to be OPERABLE to ensure safety analysis assumptions regarding RB isolation are bounded. Operability of the instrumentation includes proper operation of the sample pump. This LCO addresses only the gas sampler portion of the System.

(continued)

BASES

APPLICABILITY

The RB Purge Isolation-High Radiation instrumentation shall be OPERABLE whenever required to support OPERABILITY of the purge and mini-purge valves, per LCO 3.9.3, "Containment Penetrations." These MODES and specified conditions are indicative of those under which the potential for a fuel handling accident; and thus radiation release, is the greatest. While in MODES 5 and 6, when fuel handling in the RB is not in progress, the isolation system does not need to be OPERABLE because the potential for a radioactive release is minimal and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of 10 CFR 50.67 (Ref. 1).

ACTIONS

A.1

Condition A applies to failure of the high radiation purge isolation function when the purge and mini-purge valves are required to be OPERABLE in accordance with LCO 3.9.3, "Containment Penetrations."

With the channel inoperable during this time, the applicable Conditions and Required Actions of LCO 3.9.3 are required to be entered immediately. The immediate Completion Time is consistent with the loss of RB isolation capability under conditions in which the fuel handling accidents are possible and the high radiation function is required to provide automatic action to terminate the release.

SURVEILLANCE REQUIREMENTS

SR 3.3.15.1

This SR is the performance of the CHANNEL CHECK for the RB purge isolation-high radiation instrumentation once every 12 hours. The CHANNEL CHECK is a comparison of the parameter indicated on the radiation monitoring instrumentation channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.15.1 (continued)

something even more serious. Internal check sources may also be used to satisfy the CHANNEL CHECK requirement.

Acceptance criteria are determined by plant staff and are presented in the Surveillance Procedures. The criteria are based on a combination of the channel instrument uncertainties. The 12 hour Frequency, about once every shift, is based on operating experience that demonstrates channel failure is an unlikely event. Additionally, control room alarms and annunciators are provided to alert the operator to various "trouble" conditions associated with the instrument.

SR 3.3.15.2

This SR requires the performance of a CHANNEL FUNCTIONAL TEST once every 92 days to ensure that the channel can perform its intended function. This test verifies the capability of the instrumentation to provide the RB purge and mini-purge valve isolation on a high radiation signal.

As with any CHANNEL FUNCTIONAL TEST, this SR need not include actuation of the end devices (purge and mini-purge valves). The 92 day Frequency is based on the recommendations of NUREG-1366 (Ref. 2).

SR 3.3.15.3

CHANNEL CALIBRATION is a complete check of the instrument string including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains OPERABLE between successive tests.

The Allowable Value for the RB Purge Isolation-High Radiation is determined in accordance with the requirements of the Offsite Dose Calculation Manual (ODCM).

(continued)

BASES

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| SURVEILLANCE REQUIREMENTS | SR 3.3.15.3 (continued)<br><br>The 18 month Frequency is based on engineering judgment and Industry-accepted practice. |
| REFERENCES | 1. 10 CFR 50.67.<br><br>2. NUREG-1366, December 1992. |

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# B 3.3 INSTRUMENTATION

## B 3.3.17 Post Accident Monitoring (PAM) Instrumentation

BASES

### BACKGROUND

The function of PAM instrumentation is to display plant process variables that provide information for the operator to take the manual actions assumed for Design Basis Accident (DBA) mitigation. In addition, certain PAM instrumentation also provides information to assess the performance and status of selected plant systems following a DBA. These essential instruments, identified in FSAR, Table 7-12, (Ref. 1) address the recommendations of Regulatory Guide 1.97 (Ref. 2) as required by Supplement 1 to NUREG-0737 (Ref. 3).

The instrument channels required to be OPERABLE by this LCO are those parameters identified during the CR-3 specific implementation of Regulatory Guide 1.97 as Type A variables and non-Type A, Category I variables. Type A variables are included in this LCO because they provide the primary information that permits the control room operator to take specific manually controlled actions that are required when no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBAs, (Ref. 2). Primary information is that required for the direct accomplishment of the specified safety function; it does not include those variables associated with contingency actions. Category 1 variables are the key variables deemed risk significant for CR-3.

### APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the information is available to the control room operating staff:

a. Perform the diagnosis specified in the emergency operating procedures. These variables are restricted to pre-planned actions for the primary success path of DBAs (e.g., loss of coolant accident (LOCA));

b. Take the specified, preplanned, manually controlled actions, for which no automatic control is provided, which are required for safety systems to accomplish their safety functions;

c. Determine whether systems important to safety are performing their intended functions;

d. Determine the potential for a gross breach of the barriers to radioactivity release;

e. Determine if a gross breach of a barrier has occurred; and

f. Initiate action necessary to protect the public and estimate the magnitude of any impending threat.

(continued)

| FUNCTION | CHANNEL A | CHANNEL B |
|---|---|---|
| 15. Steam Generator Water Level (Operating Range) | **OTSG A:** SP-17-LI1 or SP-17-LIR<br>**OTSG B:** SP-21-LI1 or SP-21-LIR | **OTSG A:** SP-18-LI1<br>**OTSG B:** SP-22-LI1 |
| 16. Steam Generator Pressure | **OTSG A:** MS-106-PI1 or MS-106-PIR,<br>**OTSG B:** MS-110-PI1 or MS-110-PIR | **OTSG A:** MS-107-PI1 or MS-107-PIR<br>**OTSG B:** MS-111-PI1 or MS-111-PIR |
| 17. Emergency Feedwater Tank Level | EF-98-LI1 | EF-99-LI1 |
| 18a. Core Exit Temperature (Thermocouple) Quadrant<br>WX<br>XY<br>YZ<br>ZW | <br>IM-5G-TE/IM-6C-TE<br>IM-9E-TE/IM-13G-TE<br>IM-9H-TE/IM-100-TE<br>IM-3L-TE/IM-60-TE | <br>IM-7F-TE/IM-2G-TE<br>IM-10C-TE/IM-11G-TE<br>IM-10M-TE/IM-13L-TE<br>IM-4N-TE/IM-6L-TE |
| 18b. Core Exit Temperature (Recorder) | RC-171-TR | RC-172-TR |
| 19. Emergency Feedwater Flow | **OTSG A:** EF-25-FI1<br>**OTSG B:** EF-23-FI1 | **OTSG A:** EF-26-FI1<br>**OTSG B:** EF-24-FI1 |
| 20. Low Pressure Injection Flow | DH-1-FI1 | DH-1-FI2 |
| 21. Degrees of Subcooling | As Displayed on EMCO-38<br><br>*Note: Entry into LCO 3.3.17 is required if any of the following Hardware or RECALL Points are Out of Service.*<br><br>**Hardware**<br>Multiplexers EMCO-17/18/19<br>Comm. HUBs EMCO-07/20<br>Computers EMCO-21/40<br>Monitor EMCO-38<br>Recorder RC-171-TR<br><br>**RECALL Points**<br>RCS Pressure NR RECL-243<br>RCS Pressure WR RECL-4<br>T-Hot RECL-17/239 | As Displayed on EMCO-39<br><br>*Note: Entry into LCO 3.3.17 is required if any of the following Hardware or RECALL Points are Out of Service.*<br><br>**Hardware**<br>Multiplexers EMCO-26/27/28<br>Comm. HUBs EMCO-08/29<br>Computers EMCO-30/41<br>Monitor EMCO-39<br>Recorder RC-172-TR<br><br>**RECALL Points**<br>RCS Pressure NR RECL-40<br>RCS Pressure WR RECL-5<br>T-Hot RECL-18/240 |
| 22. Emergency Diesel Generator kW Indication | EGDG-1A Wattmeter SSF-AH Main control board indicator | EGDG-1B Wattmeter SSF-AX Main control board indicator |
| 23. LPI Pump Run Status | ESFA-LX3 (Red Light) or ESFA-HU (ES Light Matrix "A") | ESFB-LX3 (Red Light) or ESFB-HU (ES Light Matrix "B") |
| 24. DHV-42 and DHV-43 Open Position | ESFA-KN3 (Red Light) | ESFB-KN3 (Red Light) |
| 25. HPI Pump Run Status | **HPI Pump 1A:**<br>ESFA-MF7 (Red Light) or<br>ESFA-AH (ES Light Matrix "A")<br>**OR**<br>**HPI Pump 1B:**<br>ESFA-MN7 (Red Light) or<br>ESFA-AJ (ES Light Matrix "A") | **HPI Pump 1C:**<br>ESFB-MF7 (Red Light) or<br>ESFB-AH (ES Light Matrix "B")<br>**OR**<br>**HPI Pump 1B:**<br>ESFB-MV7 (Red Light) or<br>ESFB-AJ (ES Light Matrix "B") |
| 26. RCS Pressure (Low Range) | RC-147-PI1 | RC-148-PI1 |

NOTES: For Function 18a, each quadrant requires at least 2 OPERABLE detectors, one from each channel. OPERABILITY of only one detector for any quadrant constitutes entry into Condition A of LCO 3.3.17. Any quadrant with no OPERABLE detector constitutes entry into Condition C if LCO 3.3.17. Separate Condition entry is allowed for each quadrant.

For Function 25, OPERABILITY of indication is required only for the one ES selected HPI pump in each channel.

(continued)

BASES

LCO (continued)

The following list is a discussion of the specified instrument Functions listed in Table 3.3.17-1.

1. Wide Range Neutron Flux

Two wide-range neutron flux monitors are provided for post-accident reactivity monitoring over the entire range of expected conditions. Each monitor provides indication over the range of $10^{-8}$ to 100% log rated power covering the source, intermediate, and power ranges. Each monitor utilizes a fission chamber neutron detector to provide redundant main control board indication. A single channel provides recorded information in the control room. The control room indication of neutron flux is considered one of the primary indications used by the operator following an accident. Following an event the neutron flux is monitored for reactivity control. The operator ensures that the reactor trips as necessary and that emergency boration is initiated if required. Since the operator relies upon this indication in order to take specified manual action, the variable is included in this LCO. Therefore, the LCO deals specifically with this portion of the string.

2. Reactor Coolant System (RCS) Hot Leg Temperature

Two wide range resistance temperature detectors (RTD's), one per loop, provide indication of reactor coolant system hot leg temperature ($T_H$) over the range of 120° to 920°F. Each $T_H$ measurement provides an input to a control room indicator. Channel B is also recorded in the control room. Since the operator relies on the control room indication following an accident, the LCO deals specifically with this portion of the string.

$T_H$ is a Type A variable on which the operator bases manual actions required for event mitigation for which no automatic controls are provided.

Following a steam generator tube rupture, the affected steam generator is to be isolated only after $T_H$ falls below the saturation temperature corresponding to the pressure setpoint of the main steam safety valves. For event monitoring once the RCP's are tripped, $T_H$ is used along with the core exit temperatures and RCS cold leg temperature to measure the temperature rise across the core for verification of core cooling.

(continued)

# B 3.3 INSTRUMENTATION

## B 3.3.18 Remote Shutdown System

BASES

### BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation to place and maintain the plant in a safe shutdown condition from outside the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the plant in MODE 3, the Emergency Feedwater (EFW) System and the main steam safety valves or the atmospheric dump valves can be used to remove core decay heat and meet all safety requirements. The long term supply of EFW allows extended operation in MODE 3.

In the event that the control room becomes inaccessible, the operators can establish control at the remote shutdown panel and place and maintain the plant in MODE 3. Not all controls and necessary transfer switches are located at the remote shutdown panel. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations.

The OPERABILITY of the Remote Shutdown System control and instrumentation Functions ensures that there is sufficient information available on selected plant parameters to place and maintain the plant in MODE 3 should the control room become inaccessible.

### APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the unit in a safe condition in MODE 3.

The design basis for the CR-3 Remote Shutdown System is 10 CFR 50, Appendix A, GDC 19 and 10 CFR 50, Appendix R, Section L, (Ref. 1 and 2). However, the licensing basis for this LCO is limited to the manner with which FPC meets the intent of GDC 19 (i.e., FSAR Section 1.4, Criterion 11).

(continued)

BASES

| | |
|---|---|
| APPLICABLE SAFETY ANALYSES (continued) | The Remote Shutdown System was determined by the NRC to be a risk significant item required to be retained in the Technical Specifications. |

LCO

The Remote Shutdown System LCO provides the requirements for the OPERABILITY of the indication instrumentation necessary to place and maintain the plant in MODE 3 from a location other than the control room. The instrumentation required are listed in Table-3.3.18-1 in the accompanying LCO.

The instrumentation are those required for:

- Core Reactivity Control;
- RCS Pressure Control;
- RCS Temperature Control (Decay Heat Removal); and
- RCS Inventory Control.

Function of a Remote Shutdown System is OPERABLE if all instrument channels needed to support the Function are OPERABLE. Functionality of the control functions supported by the instrumentation included in this Specification is addressed outside Technical Specifications.

The Remote Shutdown System instruments covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure the Remote Shutdown System instruments will be OPERABLE if plant conditions require that the Remote Shutdown System be placed in operation.

Bases Table B 3.3.18-1 identifies the specific instrument tag numbers for the Remote Shutdown System Instrumentation listed in Table 3.3.18-1.

APPLICABILITY

The Remote Shutdown System LCO is applicable in MODES 1, 2, and 3 so that the plant can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the plant is initially subcritical and in a condition of reduced RCS energy. Under these conditions, considerabl

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.18.2

CHANNEL CALIBRATION is a complete check of the instrument loop and sensor. The SR verifies that the channel responds to the measured parameters within the necessary range and accuracy.

A Note clarifies that Function 1.a., "Reactor Trip Breaker (RTB) Position" is not required to have a CHANNEL CALIBRATION. This indication is mechanical in nature, and thus, not subject to a calibration.

The 24 month Frequency is based on the results of comprehensive instrument uncertainty calculations that accommodate 30 months of drift as approved in Amendment 152 (Ref. 3).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.
2. 10 CFR 50, Appendix R, Section L.
3. Amendment No. 152 to the CR-3 Technical Specifications, dated February 13, 1996.

Table B 3.3.18-1
Remote Shutdown System Instrumentation

| FUNCTION | INSTRUMENT NUMBER |
|---|---|
| 1. Reactivity Control | |
| a. Reactor Trip Breaker Position | CB-1<br>CB-2<br>CB-3<br>CB-4<br>A<br>B |
| b. Source Range Neutron Flux | NI-014-NI2 |
| 2. Reactor Coolant System Pressure Control | |
| a. RCS Wide Range Pressure | RC-158-PI1 OR RC-159-PI1 |
| 3. RCS Temperature Control | |
| a. RCS Hot Leg Temperature | "A" Loop: RC-4A-TI3-2<br>"B" Loop: RC-4B-TI4-2 |
| b. RCS Cold Leg Temperature | "A" Loop: RC-5A-TI2-2<br>"B" Loop: RC-5B-TI4-2 |
| c. OTSG Pressure | "A" OTSG: MS-106-PI2 OR MS-107-PI2<br>"B" OTSG: MS-110-PI2 OR MS-111-PI2 |
| d. OTSG Level | "A" OTSG Low Range Level: SP-25-LI2 OR SP-26-LI2<br>"B" OTSG Low Range Level: SP-29-LI2 OR SP-30-LI2<br>"A" OTSG High Range Level: SP-17-LI2 OR SP-18-LI2<br>"B" OTSG High Range Level: SP-21-LI2 OR SP-22-LI2 |
| e. Emergency Feedwater Flow | "A" OTSG: EF-25-FI2 OR EF-26-FI2<br>"B" OTSG: EF-23-FI2 OR EF-24-FI2 |
| f. Emergency Feedwater Tank Level | EF-98-LI2 OR EF-99-LI2 |
| 4. RCS Inventory Control | |
| a. Pressurizer Level | RC-1-LI1-2 OR RC-1-LI3-2 |
| b. High Pressure Injection Flow | A1 Injection Line: MU-23-FI8-2<br>A2 Injection Line: MU-23-FI6-2<br>B1 Injection Line: MU-23-FI5-2<br>B2 Injection Line: MU-23-FI7-2 |

# B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.12 RCS Operational LEAKAGE

BASES

BACKGROUND

During the life of the plant, the joint and valve interfaces contained in the RCS can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems. OPERABILITY of the leakage detection systems is addressed in LCO 3.4.14, "RCS Leakage Detection Instrumentation."

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting, monitoring, and quantifying reactor coolant LEAKAGE is critical. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for a LOCA in that the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes 1 gpm primary to secondary LEAKAGE as the initial condition.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The FSAR (Ref. 3) analysis for steam generator tube rupture (SGTR) assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential in terms of offsite dose.

The safety analysis for the Steam Line Break (SLB) accident assumes 1 gpm primary to secondary LEAKAGE in one generator as an initial condition (Ref. 4). The dose consequences resulting from the SLB accident meet the acceptance criteria defined in 10 CFR 50.67.

RCS operational LEAKAGE satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the reactor coolant pressure boundary (RCPB). LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment atmosphere and sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Completion Time considers the time required to complete the Action and the low probability of a second valve failing during this time period. Closing and de-activating the second valve will render the associated LPI subsystem inoperable.

B.1 and B.2

If leakage cannot be restored, or the Required Actions accomplished, the plant must be placed in a MODE in which the requirement does not apply.

To achieve this status, the plant must be placed in MODE 3 within 6 hours and in MODE 5 within 36 hours. These Required Actions will tend to reduce the leakage and also the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

Inoperability of one or more channels of ACIS renders DHV-3 or DHV-4 incapable of automatically isolating in response to a high pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the DHR system design pressure. If the ACIS is inoperable, operation may continue as long as the DHR suction penetration is isolated by at least one closed manual or deactivated automatic valve within 4 hours. This action in effect accomplishes the purpose of the autoclosure function.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Performance of leakage testing on each RCS PIV is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 5 gpm applies to each valve.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs were not individually leakage tested, one valve could have failed completely and not detected provided the other valve in series met the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

ASME, Section XI (Ref. 3) permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential). Reference 3 allows this reduced pressure testing for those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening, e.g., check valves. In such cases, the observed rate should be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

The Frequency of testing is a combination of ASME Code and PIV Order requirements.

The Inservice Testing Program implements the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 5), cold shutdown performance requirement. This requirement is based on the need to perform this Surveillance under conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the plant at power.

The Frequency of prior to entering MODE 2 whenever the plant has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months was contained in the April 20, 1981 PIV Order (Ref. 6). It was intended to provide confidence the valves re-seated following any period of extended operation with flow through the valves. The 7 day value is based on NUREG 1366 recommendations (Ref. 4).

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable RCS conditions to allow for performance of this Surveillance. The Note

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

that allows this provision is complimentary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months.

SR 3.4.13.2 and SR 3.4.13.3

Verifying ACIS is OPERABLE ensures that RCS pressure will not pressurize the DHR system beyond its design pressure of 330 psig on the suction side and 450 psig on the discharge side of the pump. The setpoint is adjusted to account for elevation differences between the pressure instrument and the drop line and is set so RCS hot leg pressure must be < 284 psig to open the valves. This setpoint ensures the DHR design pressure will not be exceeded and the DHR relief valves will not lift. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

REFERENCES

1. NUREG-75/014, Appendix V, October 1975.
2. NUREG-0677, NRC, May 1980.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWV-3423(e).
4. NUREG-1366, December 1992.
5. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWV-3422
6. NRC Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves dated 4/20/81. Includes Technical Evaluation Report, "Primary Coolant System Pressure Isolation Valves," prepared by the Franklin Research Center.

# B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.15 RCS Specific Activity

BASES

| | |
|---|---|
| BACKGROUND | The limits on specific activity ensure that the doses are within the 10 CFR 50.67 limits during analyzed transients and accidents (Ref. 1).<br><br>The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. |
| APPLICABLE SAFETY ANALYSES | The LCO limits on the specific activity of the reactor coolant ensure that the resulting doses will not exceed the 10 CFR 50.67 dose limits. These values represent a reasonable operating capability rather than a specific analytical result. RCS specific activity is an input to the dose analyses for a Steam Generator Tube Rupture, Main Steam Line Break and Letdown Line Rupture (Ref. 2).<br><br>RCS Specific Activity satisfies Criterion 2 of the NRC Policy Statement. |
| LCO | The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the primary coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides in terms of MeV). These values represent a reasonable operating capability rather than a specific analytical result.<br><br>Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an accident, lead to site boundary doses that exceed the applicable dose limits of 10 CFR 50.67. |
| APPLICABILITY | In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, the energy in the RCS is sufficient to lift secondary side relief valves in the event of a SGTR.<br><br>For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of an SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the atmospheric dump valves and main steam safety valves. |

(continued)

BASES

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate the limits of Figure 3.4.15-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling must continue for trending purposes.

The DOSE EQUIVALENT I-131 must be restored to limits within 48 hours. The Completion Time of 48 hours limits operation in the Condition, but provides a reasonable time for temporary coolant activity increases (iodine spiking or crud bursts) to be cleaned up with processing systems. As such, the Completion Time is based on engineering judgment.

The Required Actions of Condition A are modified by a Note indicating LCO 3.0.4 is not applicable. As a result, a MODE change is allowed when RCS specific activity exceeds 1.0 μCi/gm but is less than Figure 3.4.15-1. This allowance is provided because coolant cleanup activities can proceed in parallel with plant start-up.

B.1

If either Required Action and associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.15-1, the reactor must be placed in MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is required to get to MODE 3 below 500°F without challenging plant systems.

C.1 and C.2

With gross specific activity in excess of the allowed limit, an analysis must be performed within 4 hours to determine DOSE EQUIVALENT I-131. The Completion Time of 4 hours is required to obtain and analyze a sample.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

In addition, the plant must be placed in MODE 3 with RCS average temperature less than 500°F within 6 hours.

The 6 hour action to place the plant in MODE 3 with RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and the atmospheric dump valves, and prevents venting the OTSG to the environment in an SGTR event. The Completion Time of 6 hours is required to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once per 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity. The 7 day Frequency considers the unlikelihood of a gross fuel failure during that time period and is adequate based on operating history.

SR 3.4.15.2

This Surveillance is only required to be performed in MODE 1 since this is when iodine production mechanisms are large enough to yield meaningful Surveillance results. This requirement includes reactor trip events from greater than 15% rated thermal power (RTP). This ensures the iodine remains within limit during normal operation and following fast power changes when the stresses on the nuclear fuel are the greatest. The 14 day Frequency is adequate to trend changes in the iodine activity level considering gross specific activity is monitored every 7 days. The Frequency of between 2 and 6 hours after a

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.2 (continued)

change of ≥ 15% RTP within a 1 hour period is established because the iodine levels in the core peak during this time. If fuel failure were to occur, this period of time would be the most conservative time (levels would be highest) to measure iodine concentration.

A single performance of SR 3.4.15.2 can satisfy Surveillance Requirements for multiple 1 hour periods of ≥ 15% RTP power change; providing the sample is obtained and analyzed at a time that meets the 2 to 6 hour frequency for each hourly period it satisfies.

Changes of ≥ 15% RTP are changes in power of ≥ 15% RTP in one direction. A 8% RTP power increase followed by a 8% RTP power decrease is not, by itself, a ≥ 15% RTP change.

SR 3.4.15.3

SR 3.4.15.3 requires radiochemical analysis for $\bar{E}$ every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specific gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates the SR is only required to be performed 31 days after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This SR 3.0.4 type exception ensures the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10 CFR 50.67.

2. FSAR, Sections 14.2.2.1, 14.2.2.2, 14.2.2.6.

# B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

## B 3.5.2 ECCS – Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling to ensure that the reactor core is protected after any of the following accidents:

1. Loss of coolant accident (LOCA);
2. Steam generator tube rupture (SGTR); and
3. Steam line break (SLB).

There are two modes of ECCS operation: injection and recirculation. In the injection phase, all injection is initially added to the Reactor Coolant System (RCS) from the borated water storage tank (BWST). This injection flow is added via the RCS cold legs and core flood nozzles to the reactor vessel. After the BWST has been depleted to ≤ 15 feet but > 7 feet, the ECCS recirculation phase is entered as the ECCS suction is manually transferred to the reactor building emergency sump.

Two redundant, 100% capacity trains are provided. Each train consists of high pressure injection (HPI) and low pressure injection (LPI) subsystems. In MODES 1, 2, and 3, both trains must be OPERABLE. This ensures that 100% of the core cooling requirements can be provided even in the event of a single active failure.

Certain size small break LOCA scenarios require emergency feedwater to maintain steam generator cooling until core decay heat can be removed solely by ECCS cooling.

A suction header supplies water from the BWST or the reactor building emergency sump to the ECCS pumps. Separate piping supplies each train. Each HPI subsystem discharges into each of the four RCS cold legs between the reactor coolant pump and the reactor vessel. Each LPI subsystem discharges into its associated core flood nozzle on the reactor vessel and discharges into the vessel downcomer area. Control valves are set to balance the HPI flow to the RCS. This flow balance directs sufficient flow to the core to meet the analysis assumptions following a small break LOCA in one of the RCS cold legs near an HPI nozzle.

The HPI pumps are capable of discharging to the RCS at an RCS pressure above the opening setpoint of the pressurizer

(continued)

BASES

BACKGROUND (continued)

safety valves. The LPI pumps are capable of discharging to the RCS at an RCS pressure of approximately 200 psia. When the BWST has been nearly emptied, the suction for the LPI pumps is manually transferred to the reactor building emergency sump. The HPI pumps cannot take suction directly from the sump. If HPI is still needed, a cross connect from the discharge side of the LPI pump to the suction of the HPI pumps would be opened. This is known as "piggy backing" HPI to LPI, and enables continued HPI to the RCS, if needed, after the BWST is emptied to the switchover point.

In the long term cooling period, flow paths in the LPI System can be established to preclude the possibility of boric acid in the core region reaching an unacceptably high concentration. LPI can be aligned to provide for two active methods of boron dilution. In one method, the decay heat system drop line is aligned to the RB sump to allow gravity feed from the hot leg. The other method consists of aligning the Auxiliary Pressurizer Spray to provide injection of boron dilute water into the hot leg. Although important to long-term cooling, these flowpaths are not considered to be part of the primary success path for LOCA, SGTR, or SLB mitigation.

Both active methods may be affected by a failure of Engineered Safeguards Motor Control Center ES MCC 3AB. However, due to conservatisms inherent in the analyses which demonstrated the effectiveness of these methods, combined with FPC's development of restoration procedures, reasonable assurance exists that FPC can restore an active method before boron can pose a challenge to long-term core cooling. As such, an exemption from the requirements of 10CFR50 Appendix K, "Single Failure Criterion," with regard to post-LOCA boron precipitation control was granted (Ref. 7).

HPI also functions to supply borated water to the reactor core following increased heat removal events, such as large SLBs.

During a large break LOCA, RCS pressure will decrease to < 200 psia in < 20 seconds. The ECCS is actuated upon receipt of an Engineered Safeguards Actuation System (ESAS) signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately (in the programmed sequence). If offsite power is not available, the engineered safety feature (ESF) buses shed normal operating loads and are connected to the diesel generators. Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required

(continued)

BASES

LCO (continued)

Not all portions of the HPI flow path satisfy the independence criteria discussed above. Specifically, the HPI flow path downstream of the HPI/Makeup pumps is not separable into two distinct trains, and is therefore, not independent. This conclusion is based upon analysis which shows, that in the event of a postulated break in the HPI injection piping, injection flow is required through a minimum of three (3) injection legs, assuming one pump operation, or through a minimum of two (2) injection legs, assuming two HPI pump operation. When considering the impact of inoperabilities in this portion of the system, the same concept of maintaining single active failure protection must be applied. When components become inoperable, an assessment of the HPI systems ability to perform its safety function must be performed. If the system can continue to perform its safety function, without assuming a single active failure, then the 72 hour loss of redundancy ACTION is appropriate. If the inoperability renders the system, as is, incapable of performing its safety function, without postulating a single active failure, then the plant is in a condition outside the safety analysis and must enter LCO 3.0.3 immediately.

In MODES 1, 2, and 3, an ECCS train consists of an HPI subsystem and an LPI subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an ESAS signal and manually transferring suction to the reactor building emergency sump. The flowpaths used to establish either active method of boron precipitation control, although important to long-term core cooling, are not required to be OPERABLE in order to satisfy this LCO.

During an event requiring ECCS actuation, a flow path is provided to ensure an abundant supply of water from the BWST to the RCS via the HPI and LPI pumps and their respective discharge flow paths to each of the four cold leg injection nozzles and the reactor vessel. In the long term, this flow path may be manually transferred to take its supply from the reactor building emergency sump and to supply its flow to the RCS via two paths, as described in the Background section.

The flow path for each train must maintain its designed degree of independence to ensure that no single active failure can disable both ECCS trains.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the ECCS train OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The HPI pump performance is based on the small break LOCA, which establishes the pump performance curve and is less dependent on power. MODES 2 and 3 requirements are bounded by the MODE 1 analysis.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.6, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.7, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Decay Heat Removal and Coolant Circulation – High Water Level," and LCO 3.9.5, "Decay Heat Removal and Coolant Circulation – Low Water Level."

(continued)

# B 3.6 CONTAINMENT SYSTEMS

## B 3.6.1 Containment

BASES

### BACKGROUND

The containment consists of the concrete reactor building (RB), its steel liner, and the penetrations through this structure. The structure is designed to contain water and steam, as well as radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed using a three way post tensioning system. The inside surface of the containment has a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The concrete RB is required for structural integrity of the containment under DBA conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1).

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE automatic containment isolation system, or

   2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

(continued)

BASES

BACKGROUND (continued)

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks".

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the analyses of DBAs involving release of fission product radioactivity, it is assumed that the containment is OPERABLE so that the release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as $L_a$: the maximum allowable leakage rate at the calculated maximum peak containment pressure ($P_a$) resulting from the limiting DBA. The allowable leakage rate represented by $L_a$ forms the basis for the acceptance criteria imposed on all containment leakage rate testing. $L_a$ is assumed to be 0.25% of containment air weight per day in the safety analysis at $P_a$ = 54.2 psig (Ref. 3).

The dose acceptance criteria applied to accidental releases of radioactive material to the environment are given in 10 CFR 50.67 (Ref. 5).

The containment satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES

REFERENCES

1. 10 CFR 50, Appendix J, Option B
2. FSAR, Sections 14.2.2
3. FSAR, 5.2.1.1
4. 1992 Edition of the ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWE and IWL.
5. 10 CFR 50.67.
6. NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J"
7. ANSI/ANS-56.8 1994, "American National Standard for Containment System Leakage Testing Requirement"

# B 3.6 CONTAINMENT SYSTEMS

## B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder, 10 ft in diameter, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and is tested to verify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. Therefore, closure of a single door supports containment OPERABILITY. Each of the doors contain two gasketed seals and local leakage rate testing capability t ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock door is provided with limit switches that provide control room indication of door position. Additionally, control room indication is provided to alert the operator whenever an air lock door interlock mechanism is defeated.

The containment air locks form part of the containment pressure boundary. Their integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis. All leakage rate requirements are in conformance with 10 CFR 50, Appendix J, Option B (Ref. 1).

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The DBAs analyzed for dose consequences that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE so that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as $L_a$: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure ($P_a$) following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock. $L_a$ is 0.25% of containment air weight per day and $P_a$ is 54.2 psig, resulting from the limiting design basis LOCA.

The dose acceptance criteria applied to DBA releases of radioactive material to the environment are given in 10 CFR 50.67 (Ref. 4).

The containment air locks satisfy Criterion 3 of the NRC Policy Statement.

LCO

Each containment air lock forms part of the containment pressure boundary. As a part of containment, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

(continued)

BASES

LCO (continued)

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time (Ref. 5). This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component or for emergencies involving personnel safety. If the outer door is inoperable, then it may be easily accessed to repair. If the inner door is the one that is inoperable, however, then a short time exists when the containment boundary is not intact (during access through the outer door). In this context, repairs include follow-up actions to an initial failure of the air lock door seal SR in order to determine which air lock door(s) is faulty. There are circumstances where an at-power containment entry would be required during the period of time that one air lock was inoperable. In this case, entry would be made through the OPERABLE air lock if ALARA conditions permit. However, the

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BASES

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|---|---|---|
| REFERENCES | 1. | 10 CFR 50, Appendix J, Option B |
| | 2. | FSAR, Sections 14.2.2 |
| | 3. | FSAR, 5.2.1.1 |
| | 4. | 10 CFR 50.67 |
| | 5. | FSAR Section 5.2.5.2.3.1 |
| | 6. | ANSI/ANS 56.8-1994 |

BASES

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the requirements related to the control of leakage from containment during major accidents. This LCO is intended to ensure the containment leakage rates do not exceed the values assumed in the safety analysis. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analysis of any event requiring containment isolation is applicable to this LCO.

The DBAs analyzed for dose consequences that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 3). In the analysis for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential leakage paths to the environment through containment isolation valves (including containment purge valves) are minimized.

The dose acceptance criteria applied to accidental releases of radioactive material to the environment are given in 10 CFR 50.67 (Ref. 8).

The DBA analysis assumes that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, $L_a$. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times. SR 3.3.5.4 addresses the response time testing requirements.

The single-failure criterion required in the safety analyses was considered in the original design of the containment purge valves. Two valves in a series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

inboard and outboard isolation valves on each line are provided with diverse power sources, motor operated and pneumatically operated spring closed, respectively. This arrangement was designed to preclude common mode failures from disabling both valves on a purge line.

The containment purge valves may be unable to close in the environment following a LOCA. Therefore, each of the 48 inch-purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. In this case, the single-failure criterion remains applicable to the containment purge valves because of failure in the control circuit associated with each valve. Again, the 48 inch purge system valve design prevents a single failure from compromising containment OPERABILITY as long as the system is operated in accordance with the subject LCO.

The containment isolation valves satisfy Criterion 3 of the NRC Policy Statement.

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valve safety function is related to control of containment leakage rates during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 48 inch purge valves must be maintained sealed closed in MODES 1, 2, 3 and 4. The valves covered by this LCO are listed along with their associated stroke times in the FSAR (Ref. 4).

The normally closed isolation valves are considered OPERABLE when manual valves are closed, check valves have flow through the valve secured, blind flanges are in place, and closed systems are intact.

Purge valves with resilient seals must meet additional leakage rate requirements addressed as part of this Specification. All other containment isolation valve leakage rate testing is addressed by LCO 3.6.1, "Containment," as part of Type C testing.

(continued)

BASES

LCO (continued)

This LCO provides assurance that the containment isolation valves and purge valves will perform their designated safety functions to control leakage from the containment during accidents.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The following terms are defined for the purpose of implementing this Specification:

- penetration flowpath: The piping which passes through the RB liner such that a portion of the system inside the RB can communicate with the portion outside the RB. A penetration passes through the imaginary plane established by the RB liner.
- unisolated: The state of a penetration flowpath whereby the operating fluid (liquid or gas) of the system is capable of passing freely through the imaginary plane established by the RB liner.

The ACTIONS are modified by a Note allowing penetration flow paths, except for 48 inch purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow paths containing these valves may not be opened under

(continued)

BASES

ACTIONS (continued)

administrative controls. A single purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1.

Note 2 has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path.

The ACTIONS are further modified by a Note 3, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

In the event purge valve leakage results in exceeding the overall containment leakage rate, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable (except for purge valve leakage not within limit), the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration isolated in accordance with Required Action A.1, the valve used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within the 4 hour Completion Time. The specified time period is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This periodic

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.5

Verifying that the isolation time of each power operated and automatic containment isolation valve that is not locked, sealed, or otherwise secured in the isolation position is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.6

For 48 inch containment purge valves, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, is required to ensure OPERABILITY. Operating experience has demonstrated that this type of valve seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), additional purge valve testing was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 7).

The specified Frequencies are based on plant-specific as-found/as-left leakage rate data for these valves. The data indicates the CR-3 purge valve resilient seals do not degrade during the operating cycle with the valves in the sealed closed position. The 92 day Frequency after opening the valves recognizes the seals are prone to excessive leakage following use and is consistent with the NRC resolution of B-20.

A Note to this SR requires the results to be evaluated against the Containment Leakage Rate Testing Program. This ensures that excessive containment purge valve leakage is properly accounted for in determining the overall containment leakage rate to verify containment OPERABILITY.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures each automatic containment isolation valve that is not locked, sealed, or otherwise secured in the isolation position, will actuate to its isolation position on an actual or simulated actuation signal. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. (Ref. 4 and 9)

The SR is modified by a note indicating the SR is not applicable in the identified MODE. This is necessary in order to make the requirements for automatic system response consistent with those for the actuation instrumentation.

REFERENCES

1. FSAR, Section 5.3.1.
2. FSAR, Section 5.2.1.1
3. FSAR, Sections 14.2.2.
4. FSAR, Table 5-9.
5. FSAR, Section 5.3.3.1
6. Generic Issue B-24.
7. Generic Issue B-20.
8. 10 CFR 50.67.
9. FSAR, Section 5.3.2.

BASES

BACKGROUND

Containment Cooling System (continued)

Upon receipt of a high reactor building pressure ES signal (4 psig), the two operating cooling fans running at high speed will automatically stop. One cooling unit fan will automatically restart and run at low speed, provided normal or emergency power is available. In post accident operation following an actuation signal, one Containment Cooling System fan will start automatically in slow speed if one is not already running. If the lead fan fails to start or trips, a second fan will automatically start in slow speed. A fan is operated at the lower speed during accident conditions to prevent motor overload from the higher density atmosphere. The automatic changeover valves operate to provide Nuclear Service Closed Cycle Cooling (SW) System flow to the cooling units and isolate the CI System flow.

APPLICABLE SAFETY ANALYSES

The RB Spray System and Containment Cooling System limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break. The postulated DBAs are analyzed, with regard to containment ES systems, assuming the loss of one ES bus. This is the worst-case single active failure, resulting in one train of the RB Spray System and one train of the Containment Cooling System being inoperable.

The analysis and evaluation show that, under the worst-case scenario, the highest peak containment pressure is 54.2 psig (experienced during a LOCA). The analysis shows that the peak containment temperature is 278.4°F (experienced during a LOCA). Both results are less than the design values. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a power level of 2568 MWt, one RB spray train and one RB cooling train operating, and initial (pre-accident) conditions of 130°F and 17.7 psia. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

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APPLICABLE SAFETY ANALYSIS (continued)

The effect of an inadvertent RB spray actuation has also been analyzed. An inadvertent spray actuation results in a 2.5 psig containment pressure drop and is associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

The modeled RB Spray System actuation from the containment analyses is based on a response time associated with exceeding the RB pressure High-High setpoint coincident with a high pressure injection start permit actuation signal to achieve full flow through the containment spray nozzles. The Containment Spray System total response time of 90 seconds includes emergency diesel generator (EDG) startup (for loss of offsite power), block loading of equipment, spray pump startup, and spray line filling (Ref. 2).

Containment cooling train performance for post accident conditions is given in Reference 3. The result of the analysis is that one train of RB cooling will contribute sufficient peak cooling capacity during the post accident condition in conjunction with one RB spray train to successfully limit peak containment pressure and temperature to less than design values. The train post accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference 4.

The modeled Containment Cooling System actuation from the containment analysis is based on a response time associated with exceeding the containment pressure high setpoint to achieve full Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time of 25 seconds includes signal delay, EDG startup (for loss of offsite power), and service water pump startup times (Ref. 3).

The Reactor Building Spray System and the Containment Cooling System satisfy Criterion 3 of the NRC Policy Statement.

LCO

During a DBA, a minimum of one containment cooling train and one RB spray train are required to maintain the containment peak pressure and temperature below the design limits. Additionally, one RB spray train is required to remove

(continued)

BASES

LCO (continued)

iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two RB spray trains and two containment cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst-case single active failure occurs.

Each RB Spray System train includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an Engineered Safeguards Actuation System signal and manually transferring suction to the reactor building emergency sump.

Each Containment Cooling System train includes demisters, cooling coils, dampers, an axial flow fan driven by a two speed water cooled electrical motor, instruments, and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature, requiring the operation of the RB spray trains and containment cooling trains.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the RB Spray System and the Containment Cooling System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With one RB spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat

(continued)

BASES

ACTIONS

A.1 (continued)

removal capability afforded by the OPERABLE RB spray train and cooling system train(s), reasonable time for repairs, and the low probability of a DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, "Completion Times", for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

B.1 and B.2

If the inoperable RB spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time to attempt restoration of the RB spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

With one of the required containment cooling trains inoperable, the inoperable containment cooling train must be restored to OPERABLE status within 7 days. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the RB Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.2 (continued)

occurring between surveillances and has been shown to be acceptable through operating experience.

It is preferable to run the fans in slow speed for this SR since this provides additional confidence the post-accident containment cooling train circuitry is OPERABLE.

SR 3.6.6.3

Verifying that each RB spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 5). Since the RB Spray System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.4

Verifying an emergency design cooling water flow rate of ≥ 1780 gpm to any 2 of 3 containment cooling system heat exchangers (fan cooling coils) ensures the design flow rate assumed in the safety analysis is being achieved. The SR verifies that, with the SW System in the post-accident ES alignment, adequate flow is provided to the heat exchangers to remove the design basis reactor building heat load. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. While the cooling units can be aligned to the SW System during normal operations, other critical normal-running SW loads make it impractical to verify accident flow rate to the coolers with the plant on-line. On an ES actuation, these normal-running loads are isolated and the SW flow

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.4 (continued)

normally supplied to them re-directed to the post-accident loads. The 24 month Frequency was also considered acceptable based upon the existence of other Technical Specification Surveillance Requirements. A degradation in cooling unit performance between performances of this SR would likely be seen as an increase in RB temperature (monitored once per 12 hours in accordance with SR 3.6.5.1).

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification that each automatic RB spray valve that is not locked, sealed, or otherwise secured in the correct position, actuates to its correct position and that each RB spray pump starts upon receipt of an actual or simulated actuation signal. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR is modified by a note indicating the SR is not applicable in the identified MODE. This is necessary in order to make the requirements for automatic system response consistent with those for the actuation instrumentation.

SR 3.6.6.7

This SR requires verification that each required containment cooling train actuates upon receipt of an actual or simulated actuation signal. In the event of a LOCA, the air steam mixture density is much higher than normal air density. The units are not designed to handle the full flow rate at this condition. To operate the unit at full flow (motor at high speed) at this condition, will cause the motor to overload and trip. To guard the motor from overloading, the volumetric flow rate must be cut

(continued)

BASES

**APPLICABLE SAFETY ANALYSES**

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value following the accident. The LOCA radiological dose analysis assumes the amount of radioactive material available for release is reduced by operation of the RB spray system and that most of the containment volume is covered by the spray.

The analysis demonstrates that unbuffered BWST inventory, delivered by the RB Spray System during the ECCS injection phase, is adequate to remove elemental and particulate iodine from the post-LOCA containment atmosphere and limit doses to within 10 CFR 50.67 limits. The CPCS provides long-term pH control of the spray to ensure iodine does not come out of the solution and once again, become available for release.

The CPCS satisfies Criterion 3 of the NRC Policy Statement.

**LCO**

The OPERABILITY of the CPCS ensures sufficient TSP-C is maintained in the three TSP-C storage baskets to increase pH of water in the emergency sump to at least 7.0 following a LOCA. To be considered OPERABLE, the volume, density, and solubility of the TSP-C must be sufficient to raise the average spray solution pH to between 7.0 and 11.0. This pH range ensures iodine does not re-evolve from solution during the ECCS recirculation phase without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components.

**APPLICABILITY**

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the CPCS. The CPCS assists in reducing the iodine fission product inventory which re-evolves from the reactor coolant to the RB during the ECCS recirculation phase.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the CPCS is not required to be OPERABLE in MODES 5 and 6.

(continued)

BASES (continued)

ACTIONS

A.1

With the CPCS inoperable, the system must be restored to OPERABLE status within 72 hours. The pH adjustment of the Reactor Building Spray System for corrosion protection and iodine re-evolution enhancement is reduced in this Condition. The Containment Spray System would still be available and would remove iodine from the containment atmosphere in the event of a DBA. However, some of this iodine could come back out of solution without the proper long-term sump pH control. The 72 hour Completion Time takes into account the passive nature of the CPCS design and the low probability of the worst-case DBA occurring during this period.

B.1 and B.2

If the CPCS cannot be restored to OPERABLE status within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE S within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE S allows additional time for restoration of the CPCS and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

To reduce the potential for post-LOCA iodine re-evolution from the water in the sump, the containment spray must be an alkaline solution. Since the BWST contents are normally acidic, the volume of the CPCS must provide a sufficient volume of TSP-C to adjust pH for all water injected. This SR is performed to verify the availability of sufficient TSP-C volume in the three TSP-C mesh storage baskets. A volume of 246 $ft^3$ to 254 $ft^3$ of TSP-C will produce a pH range between 7.0 and 7.6 at the onset of the recirculation

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The principal function of the main steam isolation valves (MSIVs) is to isolate steam flow from the secondary side of the steam generators (OTSGs) following a steam line break (SLB). A transient such as increased steam flow through the turbine bypass valves causing low steam generator pressure would also be terminated by closure of the MSIVs.

One MSIV is located in each of four main steam lines outside, but close to, containment. The MSIVs are located downstream of the main steam safety valves (MSSVs) and steam supply lines to the emergency feedwater (EFW) pump turbine to prevent isolation of these critical steam loads in the event of MSIV closure. Closure of the MSIVs isolates the OTSGs from the turbine, turbine bypass valves, and other auxiliary steam loads.

The MSIVs are spring actuated, pneumatically-operated valves which are opened/assisted-closed by instrument air pressure (Ref. 1). These valves close on receipt of a main steam line isolation signal generated by the Emergency Feedwater Initiation and Control (EFIC) System based upon low OTSG pressure. The main steam lines can also be manually isolated from the control room.

A description of the MSIVs is contained in FSAR, Section 10.2.1.4 (Ref. 2). In isolating the main steam lines, the MSIVs satisfy 10 CFR 50 Appendix A General Design Criteria (GDC) 57 requirements for isolation of closed system lines which penetrate containment (Ref. 3).

APPLICABLE SAFETY ANALYSIS

The SLB analysis was reperformed crediting all "as built" systems and their associated response times. This is documented in FTI Summary Analysis Report 86-1266223-00, "CR-3 MSLB with MFP Trip Failure". In this analysis, EFIC isolation of Main Feedwater and Main Steam were credited. The required stroke time of the MSIVs is six seconds which includes an EFIC signal process delay and valve closure from the time of OTSG low pressure of 585 psig. The required ITS EFIC actuation on OTSG low pressure is greater than or equal to 600 psig. The lower analysis pressure is conservative.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

There are several reasons why all MSIVs are isolated on an EFIC MS isolation, including those on the intact generator. Restricting the blowdown to a single OTSG is necessary to limit the positive reactivity effects associated with the resulting Reactor Coolant System (RCS) cooldown, as well as to prevent containment overpressurization in the event of a break within the reactor building coincident with the failure of feedwater to isolate. (Ref. 4). Additionally, MSIV closure ensures that at least one OTSG remains available for RCS cooldown and capable of supplying steam to the turbine driven EFW pump.

Several SLB variations are considered in the accident analysis. Steam line isolation prevents a single break from affecting both OTSGs, allowing the unaffected OTSG to be used for RCS heat removal. A controlled cooldown can then be maintained, through operation of the EFW system and steam relief through the atmospheric dump valves or turbine bypass valves.

In the event of a single MSIV failure coincident with an SLB accident, closure of the three remaining MSIVs will prevent continued, simultaneous blowdown of both OTSGs. Thus, the accident analysis has shown the SLB can be mitigated even with the failure of a single MSIV.

In contrast with the postulated SLB events, the MSIVs are assumed to be open following a steam generator tube rupture (SGTR) accident. Following a SGTR, activity and inventory contained within the RCS is leaked into the MS System, where it is then available for release to the environment. In the evaluation of offsite dose following a SGTR, the turbine bypass valves (TBVs) were used to establish and maintain RCS cooldown, directing the leaked reactor coolant to the condenser. Within the condenser, a partial removal of iodine was considered, effectively reducing the total quantity of radioactivity contributing to the post-accident offsite dose. Although the resultant offsite dose is predicted to be considerably less than the guidelines of 10 CFR 50.67, the ability to maintain the MSIVs open is essential to keeping offsite doses within analyzed values (Ref. 5).

The MSIVs satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES

LCO

This LCO requires that all four MSIV be OPERABLE. The MSIVs are considered OPERABLE, for this Specification, when the isolation times are within limits and they close on an EFIC isolation actuation signal. Containment isolation requirements for the MSIVs are addressed in LCO 3.6.3.

MSIVs that are closed and deactivated are considered OPERABLE since they are already performing the safety function and the administrative controls to ensure function are adequate. However, the TBVs may not be OPERABLE under these circumstances. The Required Actions of LCO 3.7.4 are required to be entered in this situation if the TBV are determined to be inoperable as a result of MSIV closure and deactivation.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 50.67 limits (Ref. 6).

APPLICABILITY

The MSIVs must be OPERABLE in MODES 1, 2, and 3 since there is significant mass and energy in the RCS and OTSG and the potential for a SLB exists.

In MODES 4, 5, and 6 the pressure and temperature in the OTSGs is markedly reduced. Therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

(continued)

BASES

ACTIONS

The ACTIONS are modified by a Note which ensures appropriate remedial actions are taken in the event the turbine bypass valves, addressed in LCO 3.7.4, are rendered inoperable by a closed MSIV (MSV-411 or MSV-413).

This Note is an LCO 3.0.6 type exception, since LCO 3.0.6 would dictate the ACTIONS of Specification 3.7.4 should not be taken if the TBV is rendered inoperable solely as a result of the inoperable MSIV. In this case, it is appropriate to enter the Conditions and Required Actions of Specification 3.7.4 when one or more TBV is inoperable.

A.1 and A.2

With one or more MSIV inoperable on one OTSG, action must be taken to restore the component(s) to OPERABLE status or close the valve(s) within 8 hours. The 8 hour Completion Time is reasonable, considering the probability of an accident that would require actuation of the MSIVs occurring during this time interval. The turbine stop valves are available as backup to provide the required isolation for the majority of postulated accidents which require OTSG isolation.

Valves closed in accordance with Required Action A.1 must be verified to be closed on a periodic basis. This is necessary given the valves are not required to be deactivated, to ensure the assumptions of the safety analysis remain valid. The 7 day Completion Time is reasonable, based upon engineering judgement, in view of MSIV status indication in the control room, and other administrative controls, to ensure the valves remain in the closed position.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the Required Action and associated Completion Time are not met, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each MSIV is in accordance with the limit specified in the Inservice Testing Program. This MSIV closure time is established based upon design specifications for the valves and is consistent with the accident and containment analyses. This Surveillance is normally performed upon returning the plant to operation following a refueling outage, because the MSIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the plant generating power. As the MSIVs are not to be tested at power, they are exempt from the ASME Code, Section XI (Ref. 7) quarterly valve stroke requirements.

The Frequency for this SR is in accordance with the Inservice Testing Program and is based on the typical refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at this Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish plant conditions most representative of those under which the acceptance criterion was generated.

(continued)

BASES

REFERENCES

1. Enhanced Design Basis Document for the Main Steam System.
2. FSAR, Section 10.2.1.4.
3. 10 CFR 50, Appendix A, GDC 57.
4. FSAR, Section 14.2.2.1.7.
5. FSAR, Section 14.2.2.2.
6. 10 CFR 50.67.
7. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection, Article IWV-3400.

# B 3.7 PLANT SYSTEMS

## B 3.7.4 Turbine Bypass Valves (TBVs)

BASES

BACKGROUND

The TBVs provide a method for cooling the plant to Decay Heat Removal (DHR) System entry conditions via the main condenser. Following an accident, this is done in conjunction with the Emergency Feedwater (EFW) System, providing flow from the EFW tank (EFT-2). There are four air-operated TBVs, two per steam generator (OTSG). The TBVs are located downstream of the main steam isolation valves (MSIVs) and other remote power-operated isolation valves to permit the valves to be isolated if necessary. Each TBV is sized to pass 3.75% of rated main steam flow (418,500 lbm/hr at normal steam conditions) and combined, the valves are capable of cooling down the plant at the design rate of 100°F/hour (Ref. 1). All four TBVs are controllable from the Main Control Board as well as local manual at the valves themselves. The TBVs are not available following a loss of offsite power (LOOP) due to the loss of the Circulating Water System and eventually the condenser. However, the licensing basis for the Steam Generator Tube Rupture (SGTR) accident does not require a LOOP be assumed.

In the event of a LOOP, the Atmospheric Dump Valves (ADVs) would be relied upon to perform the secondary side heat removal function. Calculation N-00-0004 indicates that the ADVs can be used to cool down the plant and still meet 10 CFR 50.67 limits. However, the offsite dose would be significantly higher than those associated with a TBV-based cooldown.

The ADVs are air-operated valves equipped with pneumatic controllers to permit control of the cooldown rate. The valves are provided with a backup supply of bottled air. Manual valve alignment is necessary to use this air to operate the ADVs on loss of pressure in the normal instrument air supply. The air supply is sized to provide sufficient pressurized gas to operate the ADVs for four hours, the time required to cope with a Station Blackout event. This also provides the capability to operate the ADVs to minimize the radiological consequences of a Steam Generator Tube Rupture with a LOOP (a beyond design and licensing basis scenario).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSIS

The TBVs are assumed to be used by the operator to cool down the plant following the design basis SGTR event (Ref. 2). The initiating event is a double-ended rupture of a single OTSG tube, resulting in a primary to secondary leak rate of 435 gpm; too large for normal makeup to compensate. RCS pressure decreases to the Reactor Protection System (RPS) low-pressure trip setpoint and the reactor is automatically shut down. In turn, the turbine trips and the OTSGs are isolated. Prior to operator actions to cool down the unit, the TBVs, ADVs, and the main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the OTSG's pressure and temperature below the design value. This is assumed to occur over a period of one minute, 8 minutes following initiation of an event.

At 9 minutes into the event, the analysis assumes secondary side pressure has decreased to below the ADV and MSSV setpoints and that the direct release to the environment is terminated. From this point in time until 8 hours after the start of the event, both OTSGs are continuously steamed to the condenser in order to remove decay heat and cooldown/de-pressurize the RCS to DHR System entry conditions. The analysis assumes all four TBVs are available to perform this function. At 8 hours into the event, offsite radioactivity releases are terminated as DHR is assumed to be in operation.

The proper operation of the TBVs allows both OTSGs, both the faulted and intact (assumed to have a 1 gpm primary to secondary leak rate), to be steamed to the condenser. This is significant in terms of offsite dose consequences resulting from the SGTR. A gas-liquid partition factor for iodine of 1.0 E-4 was assumed for releases occurring through the condenser. Releases directly to the atmosphere assume a partition factor of 1.0. Offsite doses calculated from the event are directly proportional to the value assumed for the partition factor. Thus, proper operation of the TBVs is necessary to maintain dose consequences associated with a SGTR to a minimum.

The TBVs satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

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LCO

Each TBV (two per OTSG) is required to be OPERABLE for this LCO. Failure to meet the LCO can result in the inability to cooldown to DHR System entry conditions following a SGTR event while maintaining offsite doses to a minimum. A TBV is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and is capable of fully opening and closing when manually commanded to do so by the operator.

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APPLICABILITY

In MODES 1, 2, and 3, the pressures and temperatures in the RCS are high enough to initiate a SGTR and require secondary side depressurization. Therefore, the TBVs are required to be OPERABLE in these MODES.

In MODES 4, 5, and 6, a SGTR is not a credible event due to the reduced stresses in the generator tubes and low driving head for release to the environment.

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ACTIONS

A.1 and A.2

With one or more TBV(s) inoperable, action must be taken to restore all TBVs to OPERABLE status. The 7 day Completion Time is reasonable to repair inoperable TBVs, based on the availability of other means of depressurizing the RCS following a SGTR, and the low probability of this event occurring during the 7 day period. As an alternative to restoring the TBV(s) to OPERABLE status, the associated OTSG ADV must be verified to be OPERABLE within 7 days. This entails verifying that SR 3.7.4.1 is "current" for the ADV, or performing the Surveillance. Reliance on the ADV to satisfy the ACTIONS of this Specification is considered acceptable based on the early analysis.

B.1 and B.2

If the TBVs cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the TBVs must be able to be opened remotely and throttled through their full range. This SR ensures that the TBVs are tested through a full control cycle at least once per fuel cycle. Cycling TBVs during plant heatup satisfies this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 10.2.1.4.
2. FSAR, Section 14.2.2.2.
3. FPC Calculation N-00-0004.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

nuclear services seawater and both decay heat seawater pumps. This ensures that the required cooling capacity is provided to these systems following a steam line break, steam generator tube rupture, makeup system letdown line failure, or loss of coolant accident.

To ensure adequate heat removal capacity is available, restrictions on UHS minimum water level and maximum temperature have been specified. A minimum intake canal water level of 73.7 feet will ensure the necessary suction is available to supply the 34,900 gpm flow required to the sea water pump pit (Ref. 5). However, to ensure that operation is bounded by the probable maximum hurricane (PMH) blowout analysis, this LCO requires a minimum level of 79 feet. Additionally a maximum UHS intake canal water temperature of 95°F provides a sufficient heat sink to maintain equipment serviced by the SW and DC Systems adequately cooled (Ref. 6, 7).

The UHS satisfies Criterion 3 of the NRC Policy Statement.

LCO

The UHS is considered OPERABLE if the level and temperature are sufficient to operate the RW System for at least 30 days following the design basis LOCA without the loss of net positive suction head (NPSH) to the RW pumps, and without exceeding the maximum design temperature of the equipment served by the RW System. To meet this criteria, the UHS temperature should not exceed 95°F, and the level should not fall below 79 ft plant datum during normal unit operation.

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the UHS and is thus required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

(continued)

BASES

ACTIONS

A.1 and A.2

If the UHS is inoperable, the plant. must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

To ensure plant operation is bounded by the PMH blowout analysis, the UHS level at the intake structure must be verified to be ≥ 79 feet plant datum. The 24 hour Frequency, based on engineering judgment and industry-accepted practice, is sufficient to verify continued availability of adequate UHS level.

SR 3.7.11.2

This SR verifies that the RW System temperature is within the safety analysis assumptions related to maintaining SW and DC maximum design temperature for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water temperature is ≤ 95 °F.

REFERENCES

1. FSAR, Section 9.5.2.
2. Safety Evaluation Report (SER) for Amendment 109 dated February 14, 1989.
3. FSAR, Section 2.4.2.3.
4. Regulatory Guide 1.27.
5. Gilbert Commonwealth Report No. 2669, "Criteria for Maintaining Adequate Intake Canal Flow Capacity," revised December 2, 1987.

(continued)

## B 3.7 PLANT SYSTEMS

## B 3.7.12 Control Room Emergency Ventilation System (CREVS)

BASES

BACKGROUND

The principal function of the Control Room Emergency Ventilation System (CREVS) is to provide an enclosed environment from which the plant can be operated following an uncontrolled release of radioactivity or toxic gas.

The CREVS consists of two trains with much of the non-safety related equipment common to both trains and with two independent, redundant components supplied for major items of safety related equipment (Ref. 1). The major equipment consists of the normal duty filter banks, the emergency filters, the normal duty and emergency duty supply fans, and the return fans. The normal duty filters consist of one bank of glass fiber roughing filters. The emergency filters consist of a roughing filter similar to the normal filters, high efficiency particulate air (HEPA) filters, and activated charcoal adsorbers for removal of gaseous activity (principally iodine). The rest of the system, consisting of supply and return ductwork, dampers, and instrumentation, is not designed with redundant components. However, redundant dampers are provided for isolation of the ventilation system from the surrounding environment.

The Control Complex Habitability Envelope (CCHE) is the space within the Control Complex served by CREVS. This includes Control Complex floor elevations from 108 through 180 feet and the stair enclosure from elevation 95 to 198 feet. The elements which compromise the CCHE are walls, doors, a roof, floors, floor drains, penetration seals, and ventilation isolation dampers. Together the CCHE and CREVS provide an enclosed environment from which the plant can be operated following an uncontrolled release of radioactivity or toxic gas.

CREVS has a normal operation mode and recirculation modes. During normal operation, the system provides filtered, conditioned air to the control complex, including the controlled access area (CA) on the 95 foot elevation. When switched to the recirculation mode, isolation dampers close isolating the discharge to the controlled access area and isolating the outside air intake. In this mode the system recirculates filtered air through the CCHE.

(continued)

BASES

BACKGROUND (continued)

The control complex normal duty ventilation system is operated from the control room and runs continuously. During normal operation, the outside air intake damper is partially open, the atmospheric relief discharge damper is closed, the discharge to the CA is open, and the system return damper is throttled. This configuration allows a controlled amount of outside air to be admitted to the control complex. The design temperature maintained by the system is 75°F at a relative humidity of 50%.

Two signals will cause the system to automatically switch to the recirculation modes of operation.

1. Engineered Safeguards Actuation System (ESAS) signal (high reactor building pressure).
2. High radiation signal from the return duct radiation monitor RM-A5.

The recirculation modes isolate the CCHE from outside air to ensure a habitable environment for the safe shutdown of the plant. In these modes of operation, the controlled access area is isolated from the CCHE.

Upon detection of ESAS, the system switches to the normal recirculation mode. In this mode, dampers for the outside air intake and the exhaust to the CA will automatically close, isolating the CCHE from outside air exchange, and the system return damper will open thus allowing air in the CCHE to be recirculated. Additionally, the CA fume hood exhaust fan, CA fume hood auxiliary supply fan, and CA exhaust fan are de-energized and their corresponding isolation dampers close. The return fan, normal filters, normal fan, and the cooling (or heating) coils remain in operation in a recirculating mode.

Upon detection of high radiation by RM-A5 the system switches to the emergency recirculation mode. In this mode, the dampers that isolate the CCHE from the surroundings will automatically close. The CA fume hood exhaust fan, CA fume hood auxiliary supply fan, CA exhaust fan, normal supply fan, and return fan are tripped and their corresponding isolation dampers close. Manual action is required to restart the return fan and place the emergency fans and filters in operation. The cooling (or heating) coils remain in operation.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS

During emergency operations the design basis of the CREVS and the CCHE is to provide radiation protection to the control room operators. The limiting accident which may threaten the habitability of the control room (i.e., accidents resulting in release of airborne radioactivity) is the postulated Control Rod Ejection accident. The consequences of this event result in the limiting radiological source term for the control room habitability evaluation (Ref. 2). The CREVS and the CCHE ensures that the control room will remain habitable following all postulated design basis events, maintaining exposures to control room operators within the limits of GDC 19 of 10 CFR 50 Appendix A (Ref. 3).

The CREVS is not in the primary success path for any accident analysis. However, the Control Room Emergency Ventilation System meets Criterion 3 of the NRC Policy Statement since long term control room habitability is essential to mitigation of accidents resulting in atmospheric fission product release.

LCO

Two trains of the control room emergency ventilation system are required to be OPERABLE to ensure that at least one is available assuming a single failure disabling the other train. Failure to meet the LCO could result in the control room becoming uninhabitable in the unlikely event of an accident.

The required CREVS trains must be independent to the extent allowed by the design which provides redundant components for the major equipment as discussed in the BACKGROUND section of this bases. OPERABILITY of the CREVS requires the following as a minimum:

a. A Control Complex Emergency Duty Supply Fan is OPERABLE;

b. A Control Complex Return Fan is OPERABLE;

c. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions;

d. Ductwork and dampers are OPERABLE, and air circulation can be maintained; and

(continued)

BASES

LCO (continued)

e. The CCHE is intact as discussed below.

The CCHE boundary including the integrity of the doors, walls, roof, floors, floor drains, penetration seals, and ventilation isolation dampers must be maintained within the assumptions of the design calculations. Breaches in the CCHE must be controlled to provide assurance that the CCHE remains capable of performing its function.

If CCHE integrity cannot be maintained, the CCHE is rendered inoperable and entry into LCO Condition B is required. If the Required Action of LCO Condition B is not met within the respective Completion Time, then Condition C must be entered.

The LCO is modified by a Note allowing the CCHE boundary to be opened intermittently under administrative controls. For entry and exit through doors, the administrative control of the opening is performed by person(s) entering or exiting the area. For other designed openings such as hatches, panels and access ports, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

The ability to maintain temperature in the Control Complex is addressed in Technical Specification 3.7.18.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREVS must be OPERABLE to ensure that the CCHE will remain habitable during and following a postulated accident.

ACTIONS

A.1

With one CREVS train inoperable, action must be taken to restore the train to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the radiation protection function for control room personnel. However, the overall reliability is reduced because a failure in the OPERABLE CREVS train could result

(continued)

BASES

ACTIONS (continued)

A.1 (continued)

in loss of CREVS function. The 7 day Completion Time is based on the low probability of an accident occurring during this time period, and ability of the remaining train to provide the required capability.

B.1

With the CCHE inoperable, the CREVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE CCHE boundary within 24 hours. During the time frame that the CCHE boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radiation, toxic chemicals and smoke. Restoration of the CCHE boundary is not limited to returning the boundary to its previous condition, but can also be accomplished using temporary sealing measures as described in plant procedures and/or work instructions.

Condition B will permit maintenance and modification to the habitability envelope boundary. It also will provide the opportunity to repair the boundary in a time frame consistent with the safety significance. Breaches in the envelope, that are either planned or discovered, may be evaluated in accordance with design basis documents to determine if the CCHE remains OPERABLE. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour completion time is reasonable based on the low probability of a significant release occurring during this time and the use of compensatory measures.

C.1 and C.2

In MODE 1, 2, 3, or 4, if the inoperable CREVS train or CCHE boundary cannot be restored to OPERABLE status, within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If both CREVS trains are inoperable the CREVS may not be capable of performing the intended function and the plant is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once every month adequately checks proper function of this system. Systems such as the CR-3 design without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.12.2

This SR verifies that the required CREVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CREVS filter tests are in accordance with Regulatory Guide 1.52, (Ref. 4) as described in the VFTP Program description (FSAR, Section 9.7.4). The VFTP includes testing HEPA filter performance, charcoal absorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.12.3

This SR verifies that each CREVS train actuates to place the control complex into the emergency recirculation mode on an actual or simulated actuation signal. The Frequency of 24 months is consistent with the typical fuel cycle length.

SR 3.7.12.4

This SR verifies that CCHE integrity is maintained. The details of the program are contained in the Control Complex Habitability Envelope Integrity Program, which is required by Technical Specification 5.6.2.21. Failure to meet individual program requirements does not necessarily make the CCHE inoperable. Each individual failure should be evaluated in accordance with design basis documents to determine if the CCHE can still perform its safety function. If the CCHE can still function as required in the design basis analysis, the system remains OPERABLE.

(continued)

BASES

| | | |
|---|---|---|
| REFERENCES | 1. | FSAR, Section 9.7.2.1.g. |
| | 2. | FPC Calculation N-00-0006. |
| | 3. | 10 CFR 50, Appendix A, GDC 19. |
| | 4. | Regulatory Guide 1.52, Rev. 2, 1978. |

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B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Storage Pool Water

BASES

BACKGROUND

The water contained in the spent fuel pool provides a medium for removal of decay heat from the stored fuel elements, normally via the spent fuel cooling system. In the event fuel pool cooling is lost when the pool is 140°F, assuming a full core is discharged under the conditions of Reference 1, the pool volume provides approximately 8 hours before boiling would occur (Ref. 1). The spent fuel pool water also provides shielding to reduce the general area radiation dose during both spent fuel handling and storage.

Although maintaining adequate spent fuel pool water level is essential to both decay heat removal and shielding effectiveness, the Technical Specification minimum water level limit is based upon maintaining the pool's iodine retention effectiveness consistent with that assumed in the evaluation of a fuel handling accident (FHA). The fuel handling accident described in FSAR Section 14.2.2.3 (Ref. 2), assumes that a minimum of 23 feet of water is maintained above the stored fuel. This assumption allows the use of the pool iodine decontamination factor used in the associated offsite dose calculation.

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the FHA described in FSAR Section 14.2.2.3. The resultant 2 hour dose to a person at the exclusion area boundary and the 30 day dose at the low population zone are much less than 10 CFR 50.67 (Ref. 4) limits.

Although the water level above a damaged assembly lying on top of the fuel storage racks may be less than 23 feet, an extrapolation of the iodine removal efficiency factors indicates that the iodine removal factor used in the dose calculations will still be conversative at water levels as low as 21 feet (Ref. 5). The 23 foot criteria above the fuel in the racks will ensure at least 21 feet above the damaged assembly.

(continued)

BASES

| | |
|---|---|
| APPLICABLE SAFETY ANALYSES (continued) | Fuel storage pool water level satisfies Criterion 2 of the NRC Policy Statement. |
| LCO | The specified water level of 23 feet over the top of the irradiated fuel assemblies seated in the storage racks (156 ft plant datum) preserves the assumptions of the FHA analysis (Ref. 2). As such, it is the minimum level allowed during movement of fuel within the fuel storage pool. |
| APPLICABILITY | This LCO is only applicable during movement of irradiated fuel assemblies in the fuel storage pool. This is consistent with the safety analysis which assumes the FHA initiating event to be the drop of an irradiated fuel assembly. Control of heavy loads, i.e., damaging the fuel assembly as a result of dropping a heavy load onto it, is not addressed by the safety analysis or this Technical Specification. Plant procedures are relied upon to preven the dropping of heavy loads onto spent fuel. |

ACTIONS

<u>A.1</u>

With the fuel storage pool level less than the minimum required level, the movement of fuel assemblies in the fuel storage pool is immediately suspended. This effectively precludes the occurrence of a fuel handling accident.

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, placing the reactor in a shutdown condition in the event of an inability to suspend movement of irradiated fuel assemblies does nothing to compensate for the Required Action not met. It is inappropriate to subject the plant to a shutdown transient in this condition. In MODES 5 and 6, LCO 3.0.3 is not applicable.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

The water level in the fuel storage pool must be checked periodically. Since there is no mechanism for inadvertently lowering the level during normal operations (changes in level are procedurally controlled) and there is a low level alarm should pool level drop to approximately 24 feet above the stored fuel assemblies, a 7 day Frequency is sufficient to provide assurance of adequate water level. The Frequency is based on engineering judgment and industry-accepted practice. When refueling operations are taking place, the level in the fuel pool is at equilibrium with that in the refueling canal and in the reactor vessel. The level in the refueling canal is verified daily by the performance of SR 3.9.6.1.

REFERENCES

1. FSAR, Section 9.3.1.
2. FSAR, Section 14.2.2.3.
3. Deleted.
4. 10 CFR 50.67.
5. FPC Calculation N-00-0001.

BASES

| | |
|---|---|
| BACKGROUND (continued) | Both of the spent fuel pools are constructed of reinforced concrete and lined with stainless steel plate. They are located in the fuel handling area of the auxiliary building.<br><br>New fuel storage requirements are addressed in Section 4.0, "Design Features". |
| APPLICABLE SAFETY ANALYSES | The function of the spent fuel storage racks are to support and protect spent fuel assemblies from the time they are placed in the pool until they are shipped offsite. The spent fuel assembly storage LCO was derived from the need to establish limiting conditions on fuel storage to assure sufficient safety margin exists to prevent inadvertent criticality. The spent fuel assemblies are stored entirely underwater in a configuration that has been shown to result in a reactivity of less than or equal to 0.95 under worse case conditions (Ref. 1 and 2). The spent fuel assembly enrichment requirements in this LCO are required to ensure inadvertent criticality does not occur in the spent fuel pool.<br><br>Inadvertent criticality within the fuel storage area could result in offsite radiation doses exceeding 10 CFR 50.67 limits.<br><br>The spent fuel assembly storage satisfies Criterion 2 of the NRC Policy Statement. |
| LCO | Limits on the new and irradiated fuel assembly storage in high density racks were established to ensure the assumptions of the criticality safety analysis of the spent fuel pools is maintained.<br><br>Limits on initial fuel enrichment and burnup for both new and for spent fuel stored in pool A have been established. Two limits are defined:<br><br>1. Initial fuel enrichment must be less than or equal to 5.0 weight percent U-235, and |

(continued)

BASES

LCO (continued)

2. For new, low irradiation, and spent fuel with initial enrichment less than or equal to 5.0 weight percent and greater than or equal to 3.5 weight percent, fuel burnup must be within the limits specified in Figure 3.7.15-1. Figure 3.7.15-1 presents two areas of required fuel assembly burnup as a function of initial enrichment. For fuel with enrichment-burnup combinations in the area above the curve, there are no restrictions on where the fuel can be stored. For fuel with enrichment-burnup combinations below the curve, the fuel must be stored in a one-out-of-two checkerboard configuration with water cells that contain no fuel. The acceptability of storing this fuel in the checkerboard configuration is documented in Reference 6.

Fuel enrichment limits are based on avoiding inadvertent criticality in the spent fuel pool. The CR-3 spent fuel storage system was initially designed to a maximum enrichment of 3.5 weight percent. Enrichments of up to 5.0 weight percent are permissible for storage in spent fuel pool A a. long as the fuel burnup is sufficient to limit the worst case reactivity in the storage pool to less than or equal to 0.95. Fuel burnup reduces the reactivity of the fuel due to the accumulation of fission product poisons. Reference 1 documents that the required burnup varies linearly as a function of enrichment with 10500 megawatt days per metric ton uranium (Mwd/mtU) required for fuel with 5.0 weight percent enrichment and 0 burnup required for 3.5 weight percent enriched fuel.

Similar types of restrictions have been established for Pool B.

1. Initial fuel enrichment must be ≤ 5.0 weight percent U-235, and

2. For fuel with initial enrichment ≤ 5.0 weight percent and ≥ 2.0 weight percent, fuel burnup must be within the limits specified in Figure 3.7.15-2.

(continued)

BASES

LCO (continued)

Fuel with burnup-enrichment combinations in the area above the upper curve has no restrictions on where it can be stored. Fuel with burnup-enrichment combinations in the area between the lower and upper curves must be stored in the peripheral cells of the pool. The peripheral cells are those that are adjacent to the walls of the spent fuel pool. Fuel with burnup-enrichment combinations in the area below the lower curve cannot be stored in Pool B, but must be stored in Pool A.

The LCO allows compensatory loading techniques, specified in the FSAR and applicable fuel handling procedures, as an alternative to storing fuel assemblies in accordance with Figures 3.7.15-1 and 3.7.15-2. This is acceptable since these loading patterns assure the same degree of subcriticality within the pool.

APPLICABILITY

In general, limiting fuel enrichment of stored fuel prevents inadvertent criticality in the storage pools. Inadvertent criticality is dependent on whether fuel is stored in the pools and is completely independent of plant MODE.

Therefore, this LCO is applicable whenever any fuel assembly is stored in high density fuel storage locations.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating LCO 3.0.3 does not apply. Since the design basis accident of concern in this Specification is an inadvertent criticality, and since the possibility or consequences of this event are independent of plant MODE, there is no reason to shutdown the plant if the LCO or Required Actions cannot be met.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figure 3.7.15-1 or Figure 3.7.15-2, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance. The Immediate Completion Time underscores the necessity of restoring spent fuel pool irradiated fuel loading to within the initial assumptions of the criticality analysis.

(continued)

BASES

ACTIONS

A.1 (continued)

The ACTIONS do not specify a time limit for completing movement of the affected fuel assemblies to their correct location. This is not meant to allow an unnecessary delay in resolution, but is a reflection of the fact that the complexity of the corrective actions is unknown.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

Verification by administrative means that initial enrichment and burnup of fuel assemblies in accordance with Figure 3.7.15-1 and Figure 3.7.15-2 is required prior to storage of spent fuel in storage pool A or pool B (as applicable). This surveillance ensures that fuel enrichment limits, as specified in the criticality safety analyses (Ref. 1 and 2), are not exceeded. The surveillance Frequency (prior to storage in high density region of the fuel storage pool) is appropriate since the initial fuel enrichment and burnup cannot change after removal from the core.

REFERENCES

1. Criticality Safety Evaluation of the Pool A Spent Fuel Storage Racks in Crystal River Unit 3 with Fuel of 5.0% Enrichment, S. E. Turner, Holtec Report HI 931111, December 1993.

2. Criticality Safety Analysis of the Westinghouse Spent Fuel Storage Racks in Pool B of Crystal River Unit 3, S. E. Turner, Holtec Report HI-992128, May 1999.

3. NUREG 0800, Standard Review Plan, Section 9.1.1 and 9.1.2, Rev. 2, July 1981.

4. 10 CFR 50.67.

5. CR-3 FSAR, Section 9.6.

6. Criticality Safety Analysis of the Crystal River Unit 3 Pool A for Storage of 5% Enriched Mark B-11 Fuel in Checkerboard Arrangement With Water Holes, S. E. Turner, Holtec Report HI-992285, August 1999.

## B 3.7 PLANT SYSTEMS

## B 3.7.16 Secondary Specific Activity

BASES

BACKGROUND

Under normal operating conditions, minimal quantities of radioactive contaminants will be present in the secondary coolant due to steam generator tube or tube sheet leakage. Such leakage allows primary coolant activity to enter the steam and power conversion system, where it may be released to the atmosphere via condenser off-gas or Main Steam System leakage.

The secondary coolant is monitored and sampled to detect reactor coolant system leakage into the secondary coolant. Small amounts of leakage would be detected by this monitoring. An abnormally high specific activity is indicative of an increase in the RCS activity level or an increase in primary to secondary system leakage.

During normal operations, secondary activity must be monitored to ensure that the total annual quantity of radioactive iodine released to the atmosphere is within the requirements of 10 CFR 50, Appendix I (Ref. 1). In addition to releases of secondary coolant activity which occur during normal operations, anticipated transients or accidents which result in main steam safety valve (MSSV) or atmospheric dump valves lifting or main steam line rupture will cause direct release of secondary activity to the atmosphere.

While it is important to maintain secondary coolant specific activity within the limit to assure the offsite dose calculations are bounding for all operating conditions, the analysis also provides an early indication of increasing primary to secondary system leakage.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS

With respect to offsite dose, the limiting accident involving release of secondary coolant activity is the main steam line break (SLB) postulated as a double-ended rupture of one main steam system line between the reactor building and turbine stop valves (Ref. 3 and 4). The SLB is assumed to result in the release of the activity contained in the steam generator inventory, the activity contained in the feedwater added to the steam generator prior to isolation, and the activity added to the steam generator due to RCS leakage during the cooldown period following the steam line break. The curies of secondary (steam generator and feedwater) activity released is small in comparison to that resulting from the postulated reactor coolant leakage following the SLB. The specific activity limit on secondary coolant helps ensure this small fraction compared to primary side releases.

A complete loss of AC power and a steam generator tube rupture (SGTR) are two events that also result in offsite release of secondary coolant activity through the MSSVs and atmospheric dump valves. In the case of a complete loss of AC power, the quantity of secondary coolant released to the atmosphere could be greater than during a SLB. However, the overall offsite dose is considerably lower, since the primary to secondary leakage path will be isolated much earlier following an AC power loss than after a SLB (Ref. 6). The specific activity limit on secondary coolant helps ensure the dose from a loss of AC power will be bounded by a SLB accident.

In the case of a SGTR, the activity released from secondary side pre-break activity is insignificant compared to the activity released from the primary to secondary break flow.

(continued)

BASES

| | |
|---|---|
| LCO | The specific activity of the secondary coolant system is required to be ≤ 4.5 E-4 microcuries per gram DOSE EQUIVALENT I-131 (Ref. 7).<br><br>A secondary coolant system specific activity within this limit ensures that the offsite dose contribution of the secondary coolant activity does not exceed values considered in the safety analysis. Maintaining steam generator specific activity within this limit will ensure that the postulated post-accident doses will remain significantly less than the guideline values of 10 CFR 50.67 (Ref. 8). |
| APPLICABILITY | The limits for secondary coolant specific activity apply whenever the steam generators are required for RCS heat removal. During these conditions, the potential exists for radioactive releases to the environment via normal steam, condensate, and feedwater leakage, or as the result of a steam line failure.<br><br>In MODES 5 and 6, the steam generators are not required for RCS heat removal. Both the RCS and secondary coolant systems are depressurized, and the potential for primary to secondary leakage is minimal. |
| ACTIONS | A.1 and A.2<br><br>DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant would result in unanalyzed offsite doses in the event of an accident. Thus, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. |
| SURVEILLANCE REQUIREMENTS | SR 3.7.16.1<br><br>This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety |

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1 (continued)

analysis assumptions with respect to offsite releases. It also aids in the trending and identification of increasing isotopic concentrations that might indicate changes in reactor coolant LEAKAGE. The 31 day Frequency is based on the existence of other Surveillance Requirements to monitor activity and primary to secondary leakage rate and the existence of alarms and indications of these parameters in the control room.

REFERENCES

1. 10 CFR 50, Appendix I.
2. Deleted.
3. FSAR, Table 14-28.
4. FSAR, Section 14.2.2.
5. Deleted.
6. FSAR, Section 14.1.2.
7. FPC Fuels Calculation CMC-3, Revision 0, dated March 4, 1989.
8. 10 CFR 50.67.
9. Deleted.

B 3.7 PLANT SYSTEMS

B 3.7.18 Control Complex Cooling System

BASES

BACKGROUND

The Control Complex Cooling System provides temperature control for the control room and other portions of the Control Complex containing safety related equipment.

The Control Complex Cooling System consists of two redundant chillers, associated chilled water pumps, and parallel duct mounted air heat exchangers that can receive chilled water from either chilled water pump. A train consists of a chiller and associated chilled water pump as well as a duct mounted heat exchanger that provide cooling of recirculated control complex air. The design of the Control Complex Cooling System contains features that allow either train chiller and associated chilled water pump to provide cooling capability to either duct mounted heat exchanger. Redundant chillers and chilled water pumps are provided for suitable temperature conditions in the control complex for operating personnel and safety related control equipment. The Control Complex Cooling System maintains the nominal temperature between 70°F and 80°F.

A single chiller and associated chilled water pump will provide the required heat removal for either duct mounted heat exchanger. The Control Complex Cooling System operation to maintain control complex temperature is discussed in the FSAR, Section 9.7 (Ref. 1).

APPLICABLE SAFETY ANALYSIS

The Control Complex Cooling System consists of redundant, safety related components, with some common piping. The Control Complex Cooling System maintains the temperature between 70°F and 80°F. A single active failure of a Control Complex Cooling System component does not impair the ability of the system to perform as designed. The Control Complex Cooling System is designed in accordance with Seismic Category I requirements. The Control Complex Cooling System is capable of removing heat loads from the control room and other portions of the Control Complex containing safety related equipment, including consideration of equipment heat loads and

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

personnel occupancy requirements, to ensure equipment OPERABILITY.

The Control Complex Cooling System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two redundant trains of the Control Complex Cooling System are required to be OPERABLE to ensure that at least one train is available, assuming a single failure disables one redundant component. A Control Complex Cooling train consists of a chiller and associated chilled water pump as well as a duct mounted heat exchanger that provides cooling of recirculated control complex air. All components of an OPERABLE train must be energized by the same train electrical bus. Total system failure could cause control complex equipment to exceed its operating temperature limits. In addition, the Control Complex Cooling System must be OPERABLE to the extent that air circulation can be maintained (See Specification 3.7.12).

APPLICABILITY

In MODES 1, 2, 3, and 4, the Control Complex Cooling System must be OPERABLE to ensure that the control complex temperature will not exceed equipment OPERABILITY requirements.

ACTIONS

A.1

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy and diversity of subsystems, the inoperability of one component in a train does not render the Control Complex Cooling System incapable of performing its safety function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the Control Complex Cooling System. The intent of this Condition is to maintain a combination of equipment such that the cooling capability equivalent to 100% of a single train remains available and in operation. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

(continued)

BASES

ACTIONS

A.1 (continued)

With one or more components inoperable such that the cooling capability equivalent to a single OPERABLE train is not available, the facility is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

With one or more Control Complex Cooling trains inoperable and at least 100% cooling capability of a single OPERABLE train available, the inoperable components must be restored to OPERABLE status within 7 days*. In this Condition, the remaining Control Complex Cooling System equipment is adequate to maintain the control complex temperature. Adequate cooling capability exists when the control complex air temperature is maintained within the limits for the contained equipment and components. However, the overall reliability is reduced because additional failures could result in a loss of Control Complex Cooling System function. The 7 day Completion Time is based on the low probability of an event occurring requiring the Control Complex Cooling System and the consideration that the remaining components can provide the required capabilities.

*On a one-time basis, each Control Complex Cooling System train may be inoperable for up to 35 days to allow performance of chiller refurbishment activities. LCO 3.0.4 is not applicable during each of the one-time 35-day Completion Times. The ability to apply the one-time 35-day Completion Time to each Control Complex Cooling System train will expire on December 31, 2002.

B.1 and B.2

If the inoperable Control Complex Cooling System component cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner without challenging unit systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.18.1

Verifying that each Control Complex Cooling chiller's developed head at the flow test point is greater than or equal to the required developed head ensures that chiller's performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 3). This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

SR 3.7.18.2

This SR verifies that the heat removal capability of the system is sufficient to meet design requirements. This SR consists of a combination of testing and calculations. A 2 month Frequency is appropriate, as significant degradation of the system is slow and is not expected over this time period.

REFERENCES

1. FSAR, Section 9.7.
2. Deleted.
3. ASME, Boiler and Pressure Vessel Code, Section XI.

BASES

LCO (continued)

A sufficient lube oil supply must be available to ensure the capability to operate a single EDG at the upper limit of its 200-hour rating for 7 days. EDG lube oil sump level, in conjunction with the on-site supply and the ability to obtain replacement supplies within the required timeframe, supports the availability of EDGs required to shut down the reactor and to maintain it in a safe condition for an anticipated operational occurrence (AOO) or a postulated DBA with loss of offsite power. EDG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1,"AC Sources-Operating," and LCO 3.8.2, "AC Sources-Shutdown."

The starting air system is required to have a minimum capacity for six successive EDG start attempts without recharging the air start receivers. As such, the air start compressors are not addressed as a part of this (or any other) LCO.

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required in order to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated DBA. Since stored diesel fuel oil, lube oil, and the starting air subsystem support EDG OPERABILITY, these features are required to be within limits whenever the associated EDG is required to be OPERABLE.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 indicates separate Condition entry is allowed for each EDG. This is acceptable based upon the fact each EDG is treated as an independent entity for this Specification. Note 2 indicates LCO 3.0.4 is not applicable and MODE changes while in the ACTIONS of this Specification are permitted. It could be argued this Note is not required since this Specification allows indefinite operation. However, to avoid any future confusion on the allowance, LCO 3.0.4 has been specifically excepted. This is considered acceptable since operation in accordance with this Specification still means the EDG is OPERABLE.

(continued)

ACTIONS (continued)

A.1

With usable fuel oil volume in one or more storage tanks < 22,917 gallons, prompt action must be taken within 1 hour to verify that the combined fuel oil supply > 45,834 gallons. However, the Condition is restricted to fuel oil level reductions that maintain at least a combined 7 day supply. In this Condition, a period of 1 hour is allowed to ensure that sufficient fuel oil supply for 7 days of EDG operation at its upper 200-hour rating is available. In order to maintain the ability to treat the EDG as independent entities for the ACTIONS (from a fuel oil perspective), an artificial lower limit on stored fuel oil has been established. The minimum usable volume specified for each tank is equivalent to 3 days operation and was set to ensure a minimum combined 6 day supply.

The limit on combined supply recognizes that while one tank may contain less than 3.5 day supply, the usable volume in the other tank could be such that 7 day capacity still exists.

Verification of the fuel oil volume refers only to ascertaining the value of the total volume of the two fuel oil tanks and does not imply that the tanks must be restor to the ITS limit in one hour. If the verification shows that the combined stored volume is less than 45,834 gallons, Required Action B.1 is applicable and fuel oil level must be restored within 48 hours from the initial time of discovery.

Consistent with the Bases for Surveillance 3.0.1, OPERABILITY is verified by ensuring the associated surveillance(s) has been satisfactorily completed within the required frequency and the equipment is not otherwise known to be inoperable.

B.1

With usable fuel oil volume in one or more storage tanks < 22,917 gallons and combined fuel oil supply < 45,834 gallons, sufficient fuel oil supply for 7 days of EDG operation at its upper 200-hour rating is not available. However, the Condition is restricted to fuel oil level reductions, that maintain at least a combined 6 day supply. In this Condition, a period of 48 hours is allowed prior to declaring the associated EDG inoperable. In order to maintain the ability to treat the EDG as independent entities for the ACTIONS (from a fuel oil perspective), an artificial lower limit on stored fuel oil has been established. The minimum usable volume specified for each tank is equivalent to 3 days operation and was set to ensu a minimum combined 6 day supply.

(continued)

# B 3.8 ELECTRICAL POWER SYSTEMS

## B 3.8.4 DC Sources - Operating

BASES

BACKGROUND

The CR-3 station DC electrical power system provides the emergency diesel generator (EDG) with initial field flash and control power. It also provides both motive and control power to selected safety related equipment and is the preferred AC vital bus power (via the inverters). The design and construction of the CR-3 electrical system preceded 10 CFR 50 Appendix A. However, the general design criteria (GDCs) issued in 1971 were considered in the design and construction. The electric power system for CR-3 is in compliance with the intent of 10 CFR 50 Appendix A, GDC 17, Electric Power Systems (Ref. 1), in that it is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

The 250/125 VDC electrical power system consists of two independent and redundant safety related Class 1E DC electrical power subsystems (Train A and Train B). Each subsystem consists of two 125 VDC batteries, the associated battery charger for each battery, and all the associated control equipment and interconnecting cabling.

The 250 VDC source is obtained by use of two 125 VDC batteries connected in series. Additionally, there is one spare battery charger per subsystem, which provides backup service in the event that one of the normally aligned battery chargers is out of service. The spare battery charger meets the requirements for independence and redundancy between subsystems and as such, may be substituted for one of the preferred chargers for the purposes of satisfying this LCO.

During normal operation, the 250/125 VDC load is powered from the battery chargers with the batteries floating on the system. In the event of a loss of normal power to the battery charger, the DC loads are automatically powered from the station 1E batteries.

(continued)

BACKGROUND (continued)

The Train A and Train B DC electrical power subsystems provide the control power for their associated Class 1E AC power load group, 4160 V switchgear, and 480 V load centers. The DC electrical power subsystems also provide DC electrical power to the inverters, which in turn power the AC vital buses.

The DC power distribution system is described in more detail in the Bases for LCO 3.8.9, "Distributions System-Operating," and LCO 3.8.10, "Distribution Systems-Shutdown."

Each battery has adequate storage capacity to carry the required loads continuously for at least 2 hours and to perform three complete cycles of intermittent loads discussed in the FSAR, Chapter 8 (Ref. 4).

Each 250/125 VDC battery is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem is located in an area separated physically and electrically from the other subsystem to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. There is no shared equipment between redundant Class 1E subsystems, such as batteries, battery chargers, or distribution panels.

The batteries are rated at 1708 amp-hours. This time-current capacity is based on the discharge of 116 cells from the fully charged condition down to 1.81 volts per cell (average of all cells) at maximum discharge. The minimum cell voltage requirement corresponds to a total minimum output of 105 volts per battery bank at 77°F (Ref. 4).

The Train A and Train B DC electrical power subsystems have ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads discussed in the FSAR, Chapter 8 (Ref. 4).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 6) and Chapter 14 (Ref. 7), assume that Engineered Safeguards (ES) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the EDGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC electrical power subsystems is consistent with the initial assumptions of the accident analyses and the design basis of the plant. This includes maintaining at least one DC electrical power subsystem OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power; and

b. A worst-case single failure.

DC electric power subsystems satisfy Criterion 3 of the NRC Policy Statement.

LCO

Two DC electric power subsystems are required to be OPERABLE in order to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Each DC electrical power subsystem (i.e. train) consists of two batteries, a battery charger for each battery and the corresponding control equipment and interconnecting cabling within the train. Loss of one DC electrical power subsystem (train) does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE DC electrical power subsystem requires both batteries and their respective required chargers to be operating and connected to the associated DC buses.

APPLICABILITY

Two DC electrical power subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

(continued)

BASES

APPLICABILITY (continued)

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

DC electrical power requirements for MODES 5 and 6 are addressed in the Bases for LCO 3.8.5, "DC Sources-Shutdown."

ACTIONS

A.1

In Condition A, one safety system train is no longer capable of completely responding to an event (single failure protection is lost). Additionally, there is an increased potential for a loss of the associated DC buses during normal operation. It is therefore imperative that the operator's attention focus on stabilizing the plant in order to minimize the potential for complete loss of DC power to the affected train.

If one of the required DC electrical power subsystems is inoperable (e.g., inoperable battery, inoperable required battery charger(s), or inoperable battery charger and associated inoperable battery), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. However, a subsequent worst-case single failure would result in the complete loss of the remaining 250/125 VDC electrical power subsystems with attendant loss of ES functions. Therefore, continued power operation is limited to 2 hours. The 2 hour Completion Time is based on the recommendations of Regulatory Guide 1.93 (Ref. 8) and reflects a reasonable time to assess plant status as a result of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare for a safe and orderly plant shutdown.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8

A battery performance discharge test is a test of the constant current capacity of a battery to detect any change in the capacity determined by the acceptance test. The performance test is normally done in the as-found condition after the battery has been in service for a period of time. The test is intended to determine overall battery degradation due to age and usage.

A battery modified performance discharge test is described in the bases for SR 3.8.4.7. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.8; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.8 while satisfying the requirements of SR 3.8.4.7 at the same time.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 9) and IEEE-485 (Ref. 5). These references recommend that consideration be given to replacing the battery if its capacity is below 80% of the manufacturer rating. A capacity of 80% is an indication that the battery rate of deterioration is potentially increasing, even if there is still ample capacity to meet the design load requirements.

The Frequency for this test is 60 months, or more frequently as the battery approaches the end of its expected life, or other signs of degradation are present. A 12 month Frequency is established if the battery shows degradation or has reached 85% of its expected life with capacity < 100% of manufacturer's rating. Degradation is indicated, according to IEEE-450 (Ref. 9), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is below the manufacturer rating. A 24 month Frequency is established when the battery reaches 85% of the expected life, but the capacity is still ≥ 100% of the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 9).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.8 (continued)

This SR is modified by a Note indicating the SR should not be performed in MODES 1, 2, 3, or 4 since performing the Surveillance would perturb the electrical distribution system and challenge safety systems. However, the Note acknowledges that should an unplanned event occur in MODES 1, 2, 3, or 4, that following verification that the acceptance criteria of the SR are met, the event can be credited as a successful performance of this SR.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, March 10, 1971.
3. Proposed IEEE-308, dated 1969.
4. FSAR, Chapter 8.
5. IEEE-485-1983, June 1983.
6. FSAR, Chapter 6.
7. FSAR, Chapter 14.
8. Regulatory Guide 1.93, December 1974.
9. IEEE-450-1995.
10. Regulatory Guide 1.32, February 1977.
11. Regulatory Guide 1.129, December 1974.
12. CR-3 Calculation E90-0099.

## B 3.9 REFUELING OPERATIONS

### B 3.9.2 Nuclear Instrumentation

BASES

**BACKGROUND**

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation (NI) System. These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The installed source range neutron flux monitors are fission chambers or are boron tri-fluoride (BF3) detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers seven decades of neutron flux (0.1 to 1E+6 cps). The instruments provide continuous visual indication in the control room and an audible indication to alert operators to a possible reactivity excursion. Audible indication is also required within containment to alert personnel working on the refueling floor to an increasing count rate. NI System design criteria is contained in Reference 1.

**APPLICABLE SAFETY ANALYSES**

Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity, such as by a boron dilution accident or an improperly loaded fuel assembly. The safety analysis of the uncontrolled boron dilution accident is described in Reference 2. While the analysis does not identify the specific indication instrumentation assumed to alert the operator to the event, it is reasonable to assume the source range neutron flux monitors are a primary indication for a wide range of postulated dilution sources. The analysis of the most limiting uncontrolled boron dilution accident demonstrates that there is sufficient time and indication available for the operator to detect the event and take corrective action to terminate the event prior to reaching criticality.

The source range neutron flux monitors satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES

LCO

This LCO requires two source range neutron flux monitors OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. This LCO is met by use of either the BF3 detectors, or the post-accident monitoring wide range neutron flux instrumentation, since they have been shown to be functionally equivalent. The use of portable detectors is also permitted for purposes of complying with this LCO. If used, portable detectors should be functionally equivalent to the installed source range monitors and satisfy the applicable Surveillance Requirements.

Audible indication from one of the neutron monitors is required in both the control room and containment in order to alert personnel to changes in count rate.

APPLICABILITY

In MODE 6, the source range neutron flux monitors are required to be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels. In portions of MODE 2, MODE 3, 4 and 5, the monitors are required to be OPERABLE by LCO 3.3.9, "Source Range Neutron Flux."

In MODE 1, the neutron flux level is above the indicated range of the monitors, the BF3 monitors are de-energized and no longer relied upon for reactivity and power level monitoring. Thus, there are no requirements on source range neutron flux monitors in MODE 1.

ACTIONS

A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundant neutron count rate indication has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position, but it does preclude choosing a position that would result in a positive reactivity addition to the core.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

With no source range neutron flux monitor OPERABLE, actions to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, actions shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

With no source range neutron flux monitor OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are suspended (in accordance with Required Actions A.1 and A.2), the core reactivity condition should remain relatively stable until two source range neutron flux monitors are restored to OPERABLE status. Additional confidence in the plant's stability during this condition is provided by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and provides prompt confirmation the plant is operating in an acceptable condition. The subsequent Completion Time of once per 12 hours ensures that unplanned changes in boron concentration would be identified. The Completion Time is reasonable based on engineering judgment, considering the low probability of a change in core reactivity during this time period.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK. It is based on the assumption that the two indication channels should be consistent with each other given existing core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

The Frequency of 12 hours is based on engineering judgment and is consistent with the Frequency specified for the equivalent SR of the same instruments in LCO 3.3.9.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION every 24 months. The CHANNEL CALIBRATION for the source range nuclear instrumentation is a complete check and re-adjustment of the channels, from the count rate amplifier input to the indicators. The 24 month Frequency is based on the results of comprehensive instrument uncertainty calculations that accommodate 30 months of drift as approved in Amendment 152 (Ref. 3).

Performance of SR 3.3.17.2 or SR 3.3.9.2 meets the requirements of this Surveillance.

There is a preamplifier in the instrumentation loop. However, there are no calibration adjustments possible on the preamplifier. The preamplifier has a fixed gain which is set by means of a physical link held in place by screws and which is therefore not subject to unintentional misadjustment. Proper functioning of the preamplifier is initially verified after it or the detector is replaced and is periodically verified by the routine surveillance CHANNEL CHECK required by SR 3.9.2.1. A malfunction or failure of the preamplifier will be apparent as a change in channel output and will result in preamplifier replacement.

This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. The detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output.

REFERENCES

1. FSAR, Section 7.3.1.2.
2. FSAR, Section 14.1.2.4.
3. Amendment No. 152 to the CR-3 Technical Specifications, dated February 13, 1996.

# B 3.9 REFUELING OPERATIONS

## B 3.9.3 Containment Penetrations

BASES

BACKGROUND

An accident which occurs during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment will have any released radioactivity limited from escaping to the environment. In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, the requirement to isolate the containment from the outside atmosphere is less stringent than those established for MODES 1 through 4. In order to make this distinction, the penetration requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths for radioactivity are closed or capable of being closed.

The containment equipment hatch or outage equipment hatch (OEH) provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch or OEH must be held in place by at least four bolts. The required number of bolts is based on dead weight and is acceptable due to the low likelihood of a pressurization event. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door in the OEH (if installed) must always remain closed or be capable of being closed.

The containment air locks provide a means for personnel access during MODES 1, 2, 3, and 4 in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. However, during periods of unit shutdown when containment OPERABILITY is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment ingress and egress is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed or be capable of being closed.

(continued)

BASES

BACKGROUND (continued)

If the door in the OEH (if installed) or both doors in the containment air locks are open when containment closure is required, a designated individual must be readily available to close the door in the OEH and at least one door in each air lock. Operations personnel directly involved in refueling operations shall be aware of the identity of the designated individual(s). The designated individual(s) shall remain within sufficient proximity to the open doors to assist in evacuation of personnel inside containment and to close the open door(s) as soon as evacuation is completed.

The requirements on containment penetration closure ensure that a release of fission product radioactivity to the environment from the containment will be limited. The closure restrictions are sufficient to limit fission product radioactivity release from containment due to a fuel handling accident during refueling.

In MODE 6, it is necessary to periodically recirculate/exchange RB atmosphere in order to minimize radiation uptake during the conduct of refueling operations. The 48 inch purge valves are normally used for this purpose, but the mini-purge valves may be relied upon as well. Both valve types are automatically isolated on a unit vent-high radiation signal (from RMA-1). So long as one valve in the flow path is OPERABLE, these lines may remain unisolated during the subject plant conditions.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated by a minimum of one isolation device. Isolation may be achieved by an automatic or manual isolation valve, blind flange, or equivalent. Equivalent isolation methods include use of a material (e.g., temporary sealant) that can provide a temporary, atmospheric pressure ventilation barrier for the other containment penetrations during fuel movements.

These penetrations may be open provided the total calculated flow rate out the open penetrations is less than or equal to the equivalent flow rate through a 48 inch containment purge line penetration. This allowance is consistent with the CR-3 fuel handling accident inside the reactor building. The licensing basis analysis assumed a puff release of radionuclides from the RB following the FHA event. No credit was taken for the RB purge filters. Limiting the flow rate out the open penetrations to a flow rate less than or equal to the flow rate through the RB purge system is reasonable and conservative, given the plant licensing basis. Offsite doses from this analysis are within 10 CFR 50.67 limits. With the containment purge valves OPERABLE, no leakage value has to be assigned to these penetrations, and the entire

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BASES

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|---|---|
| BACKGROUND (continued) | 50,000 cfm can be allocated to other penetrations providing direct access. With the containment purge valves inoperable, these valves are allowed to be open during the Applicability of this Specification, however; no additional penetrations are allowed to be un-isolated during this time. |
| APPLICABLE SAFETY ANALYSES | During CORE ALTERATIONS or movement of fuel assemblies within containment with irradiated fuel in containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Canal Water Level," the administrative limit on minimum decay time of 72 hours prior to the movement of irradiated fuel in the vessel, and this LCO ensure that the release of fission product radioactivity subsequent to a fuel handling accident results in doses that are within the requirements specified in 10 CFR 50.67.<br><br>Containment penetrations satisfy Criterion 3 of the NRC Policy Statement. |
| LCO | This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity from containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for penetrations containing an OPERABLE purge or mini-purge valve. For the containment air locks and the OEH (if installed), both doors in the air locks and the door in the OEH may be open only under administrative controls. For the containment purge and mini-purge valves to be considered OPERABLE, these valves (penetrations) must be automatically isolable on a unit vent-high radiation isolation signal.<br><br>The definition of "direct access from the containment atmosphere to the outside atmosphere" is any path that would allow for transport of containment atmosphere to any atmosphere located outside the containment structure. This includes the Auxiliary Building. As a general rule, closed or pressurized systems do not constitute a direct path |

(continued)

BASES

LCO (continued)

between the RB and outside environments. All permanent and temporary penetration closures should be evaluated to assess the possibility for a release path to the outside environment. For the purpose of determining what constitutes a "direct access" path, no failure mechanisms should be applied to create a scenario which results in a "direct access" path. For example, line breaks, valve failures, power losses or natural phenomenon should not be postulated as part of the evaluation process.

These penetrations may be open provided the total calculated flow rate out the open penetrations is less than or equal to the equivalent flow rate through a 48 inch containment purge line penetration.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is the period of highest risk potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment." When CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted; the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

With the containment equipment hatch, OEH, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not in the required status, including failure to implement required administrative controls for open OEH and air lock doors and the containment purge or mini-purge valve penetrations not capable of automatic isolation when the penetrations are unisolated, the plant must be placed in a condition in which the isolation function is not

(continued)

## B 3.9 REFUELING OPERATIONS

### B 3.9.6 Refueling Canal Water Level

BASES

**BACKGROUND**

The movement of irradiated fuel assemblies within containment or performance of CORE ALTERATIONS requires a minimum refueling canal water level of 156 ft plant datum. This maintains sufficient water level above the fuel contained in the vessel and the bottom of the fuel transfer canal, and the spent fuel pool to ensure iodine fission product activity is retained in the water to a level consistent with the dose analysis of a fuel handling accident (Ref. 4). Sufficient iodine activity would be retained to limit offsite doses from the accident to well within 10 CFR 50.67 limits (Ref. 3).

**APPLICABLE SAFETY ANALYSES**

During CORE ALTERATIONS and movement of irradiated fuel assemblies, the water level in the refueling canal is an assumed initial condition in the analysis of the fuel handling accident in containment. This relates to the assumption that 99% of the total iodine released from the fuel is retained by the refueling canal water. There are postulated drop scenarios where there is < 23 ft above the top of the fuel bundle and the surface. In particular, this is the case for the period of time during which the assembly travels between the cavity and the deep end of the refueling canal. During this time, there is potentially 21 feet of water between the reactor vessel flange (135 ft plant datum) and the surface of the pool. The iodine retention factors used in the dose assessment are still conservative at water levels of 21 feet above the damaged fuel (Ref. 4). The 156 ft value was chosen to be consistent with the level specified for LCO 3.7.13, "Fuel Storage Pool Water Level" and plant configuration.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The fuel handling accident analysis inside containment is described in Reference 4. With a minimum water level of 23 ft above the stored fuel, and the administrative limit on minimum decay time of 72 hours prior to movement of irradiated fuel in the vessel, analyses demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water such that offsite doses are maintained within allowable limits (Ref. 3).

Refueling canal water level satisfies Criterion 2 of the NRC Policy Statement.

LCO

A minimum refueling canal water level of 156 ft plant datum is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits. This minimum level also ensures an adequate operational window between the surface of the pool and the transfer winch for the RB fuel handling equipment.

APPLICABILITY

This Specification is applicable during CORE ALTERATIONS and when moving irradiated fuel assemblies within the containment. The LCO minimizes the potential of a fuel handling accident in containment which results in offsite doses greater than those calculated by the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Water level requirements for fuel handling accidents postulated to occur in the spent fuel pool are addressed by LCO 3.7.13, "Fuel Storage Pool Water Level."

ACTIONS

A.1, A.2 and A.3

With a refueling canal water level of < 156 ft plant datum, all CORE ALTERATIONS or movement of irradiated fuel assemblies shall be suspended immediately to preclude a fuel handling accident from occurring. The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

(continued)

BASES

ACTIONS

A.1, A.2 and A.3 (continued)

In addition to immediately suspending CORE ALTERATIONS or movement of irradiated fuel, actions to restore refueling canal water level must be initiated immediately. The immediate Completion Time is based on engineering judgment. When increasing refueling canal water level the boron concentration of the make-up and the effect of this concentration on the minimum specified in the COLR (Ref. LCO 3.9.1) must be considered.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum refueling canal water level of 156 ft plant datum ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are assumed to result from a postulated fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Deleted.

2. FSAR Section 14.2.2.3.

3. 10 CFR 50.67.

4. FPC Calculation N-00-0001.