DUKE ENERGY.

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

Docket 50-302
Operating License No. DPR-72

ITS 5.6.2.17

May 27, 2015
3F0515-05

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

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

References:
1. NRC to CR-3 letter dated March 13, 2013, "Crystal River Unit 3 Nuclear Generating Plant Certification of Permanent Cessation of Operation and Permanent Removal of Fuel from the Reactor" (ADAMS Accession No. ML13058A30)
2. CR-3 to NRC letter dated October 29, 2013, "Crystal River Unit 3 – License Amendment #316, Revision 0, Revise and Remove License Conditions and Revision to Improved Technical Specifications to Establish Permanently Defueled Technical Specifications" (ADAMS Accession No. ML13316C083)

Dear Sir:

Duke Energy Florida, Inc. hereby provides 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 CR-3 ITS.

CR-3 ITS Section 5.6.2.17 requires that changes made without prior NRC approval be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). Due to an administrative error, the CR-3 ITS Bases changes from 2010 through 2014 were not submitted. This discovery has been entered into the CR-3 Corrective Action Program.

CR-3 has been shut down since September 2009, when a planned steam generator replacement outage began. During the 2010 through 2014 timeframe, conditions at CR-3 were such that the plant was not in any ITS MODE of Applicability. In Reference 1, the NRC acknowledged CR-3's certification of permanent cessation of power operation and permanent removal of fuel from the reactor vessel.

In Reference 2, CR-3 submitted a License Amendment Request (LAR), currently under NRC review, which proposes to extensively revise the CR-3 ITS in order to create the CR-3 Permanently Defueled Technical Specifications (PDTS). The PDTS LAR will remove the Technical Specification Bases that were affected as part of the aforementioned administrative error.

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

A001
NRR

This letter establishes no new regulatory commitments.

If you have any questions regarding this submittal, please contact Mr. Phil Rose, Lead Engineer, Nuclear Regulatory Affairs, at (352) 563-4883.

Sincerely,

Phyllis A Dixon

Phyllis A. Dixon
Director – Decommissioning Support
Crystal River Nuclear Plant

/ff

Attachments:

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

xc: NRR Project Manager
Regional Administrator, Region I

DUKE ENERGY FLORIDA, INC.

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT A

INSTRUCTIONS FOR UPDATING THE CRYSTAL RIVER UNIT 3 ITS BASES

INSTRUCTIONS FOR UPDATING
THE CRYSTAL RIVER UNIT 3
IMPROVED TECHNICAL SPECIFICATIONS

5/19/15

Page(s) to be Removed	Page(s) to be Added	
Page(s)	Pages(s)	Revision
ITS LCO LOEPages (1-4)	ITS LCO LOEPages (1-4)	5/19/15
ITS Bases LOEPages (1-8)	ITS Bases LOEPages (1-8)	5/19/15

NOTE – PassPort should reflect ITS Amendment 246 and ITS Bases Revision 88.

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DUKE ENERGY FLORIDA, INC.

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ATTACHMENT B

CRYSTAL RIVER UNIT 3 LIST OF EFFECTIVE PAGES
(ITS AND ITS BASES)

IMPROVED TECHNICAL SPECIFICATIONS

List of Effective Pages
(Through Amendment 246 and ITS Bases Revision 88)

Amendment Nos. 159, 164, 166, 171, 173, 181, 189, 190, 226, 235, 238, 239, 242, 243, 245 and 246 amended the CR-3 Operating License, only, and did not effect changes to the ITS LCOs or Bases.

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DUKE ENERGY FLORIDA, INC.

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT C

REPLACEMENT CRYSTAL RIVER UNIT 3 ITS BASES PAGES

BASES

BACKGROUND

b. OTSG Level -Low (continued)

The signals are also used by EFIC after EFW has been actuated to control OTSG level at the low level setpoint of a nominal 32 inches above the lower tubesheet when one or more RCPs are operational.

The lower and upper taps for the low range level transmitters are located at a nominal 8 inches and 279 inches, respectively, above the upper face of the OTSG's lower tube-sheet. The string is calibrated such that only the first 150 inches of indication are used. OTSG Level -Low was chosen as an EFW automatic initiation parameter because it represents a condition where feedwater is insufficient to meet the primary heat removal requirements and additional cooling water is necessary.

c. OTSG Pressure-Low

Four transmitters associated with each OTSG provide the EFIC System with channels A through D of OTSG Pressure-Low. These same transmitters provide input signals to EFIC MFW and Main Steam Line Isolation Functions. When OTSG pressure drops below the bistable setpoint of 600 psig on a given channel, an EFW Initiation signal is sent to both trains of automatic actuation logic. The low pressure Function may be manually bypassed when pressure in either OTSG is less than 750 psig. The EFIC channel bypass is automatically removed when both OTSGs outlet pressure increases above 750 psig. The low pressure operational bypass allows for normal cooldown without EFIC actuation.

OTSG Pressure-Low is a primary indication and actuation signal for steam line breaks (SLBs) or feedwater line breaks. For small breaks, which do not depressurize the OTSG or take a long time to depressurize, automatic actuation is not required. The operator has time to diagnose the problem and take the appropriate actions.

(continued)

BASES

BACKGROUND (continued)

d. RCP Status

A loss of power to all four RCPs is an immediate indication of a pending loss of forced flow in the Reactor Coolant System. The RPS acts as the sensor for this EFIC Function by providing a loss of RCP indication for each pump to each EFIC channel.

When a minimum of two EFIC channels recognize the loss of all RCPs, EFIC will automatically actuate EFW and control level to natural circulation value in the OTSG. This higher setpoint provides a thermal center in the OTSG at a higher elevation than that of the reactor to ensure natural circulation as long as adequate subcooling margin is maintained.

To allow RCS heatup and cooldown without actuation, a bypass permissive of 10% RTP is used. The 10% bypass permissive was chosen because it was an available, qualified Class 1E signal at the time the EFIC System was designed. When the first RCP is started, the "loss of four RCPs" initiation signal may be manually reset. If the bypass is not manually reset, it will be automatically reset at 10% RTP. During cooldown, the bypass may be inserted at any time THERMAL POWER has been reduced below 10%. However, for most operating conditions, it is recommended that this trip function remain active until after the Decay Heat Removal System has been placed in operation and just prior to tripping the last RCP. This trip function must be bypassed prior to stopping the last RCP in order to avoid an EFW actuation.

(continued)

BASES

BACKGROUND

Bypass (continued)

EFIC channel maintenance bypass does not bypass EFW Initiation from Engineered Safeguards Actuation System (ESAS) Channel A and Channel B high pressure injection (HPI) actuation. However, the EFIC initiation on HPI actuation is bypassed when ESAS is bypassed.

The operational bypass provisions were discussed as part of the individual Functions described earlier.

3, 4. Main Steam Line and MFW Isolation

FSAR Figure 7-26, (Ref. 3) illustrates one channel of the EFIC Main Steam Line and MFW Isolation logic. Four pressure transmitters per OTSG provide EFIC with channels A through D of OTSG pressure. The description of the channels was described earlier for EFW Initiation.

Once isolated, manual action is required to defeat the isolation command if desired. The EFIC System is designed to perform its intended function with one channel in maintenance bypass (in effect, inoperable) and a single failure in one of the remaining channels. This design complies with IEEE-279-1971 (Ref. 4) due to the redundancy and independence in the EFIC design.

APPLICABLE SAFETY ANALYSES

1. EFW Initiation

The DBA which forms the basis for initiation of EFW is a loss of MFW transient. In the analysis of this transient, SG Level -Low is the parameter assumed to automatically initiate EFW. Although loss of both MFW pumps is a direct and immediate indicator of loss of MFW, there are other scenarios (such as valve closure) that could potentially cause a loss of feedwater. Therefore, the loss of MFW analysis conservatively assumed EFW actuation on low OTSG level. This assumption yields-the minimum OTSG inventory available for heat removal and is, therefore, conservative for

(continued)

BASES

APPLICABLE SAFETY ANALYSES

1. EFW Initiation (continued)

evaluation of this event. If the loss of feedwater is a direct result of a loss of the MFW pumps, EFW will be actuated much earlier than assumed in the analysis. This would increase OTSG heat transfer capability sooner in the event and would lessen the severity of the transient.

OTSG Pressure-Low is a primary indication and provides the actuation signal for SLBs or MFW line breaks. Only one of the four SLB cases examined in the FSAR assumes normal automatic actuation of EFW. The other three cases assume manual initiation after 15 minutes. For small breaks, which do not depressurize the OTSG or take a long time to depressurize, automatic actuation is not required. The operator has sufficient time to diagnose the problem and take the appropriate actions.

Loss of four RCPS is a primary indicator of the need for EFW in the Safety analyses for loss of electric power and loss of coolant flow.

2. EFW Vector Valve Control

The SLB analysis was re-performed crediting all "as built" systems and their associated response times. This is documented in AREVA Engineering Information Record 51-9108715-000, "CR-3 ROTSG Support - Limited Scope DOE." In this analysis, EFIC isolation of Main Feedwater and Main Steam were credited. The Feed Only Good Generator (FOGG) logic was also credited for termination of an overfeed condition for either OTSG during postulated DC power failures.

3, 4. Main Steam Line and MFW Isolation

The SLB analysis was re-performed crediting all "as built" if systems and their associated response times. This is documented in AREVA Engineering Information Record 51-9108715-000, "CR-3 ROTSG Support - Limited Scope DOE." In this analysis, EFIC isolation of Main Feedwater and Main Steam were credited. The worst case evaluated single failure of the feedwater isolation system was determined to be a failure of the MFP to trip. This accident is terminated by the closing of the MFP suction valves and the downstream block valves are not credited because the block valves can not close

(continued)

BASES

APPLICABLE SAFETY ANALYSES

3, 4. Main Steam Line and MFW Isolation (continued)

against MFP discharge pressure. However, because of the slower closing low load block valves, the startup and main block can close because the pressure drop is occurringacross the slower closing low load block valves. Once the suction valves close, terminating feedwater flow, the slower closing low load block valves will close. The single failure of the suction valve failing to close, with a MFP trip and reliance upon the slower closing low load block valves for accident termination, is bounded by the MFP failing to trip. The mass and energy release for the MFP failing to trip is the bounding accident response for a SLB.

The EFIC System satisfies Criterion 3 of the NRC Policy Statement.

LCO

All instrumentation performing an EFIC System Function listed in Table B 3.3.11-1 shall be OPERABLE. Four channels are required OPERABLE for all EFIC instrumentation channels to ensure that no single failure prevents actuation of a train. Each EFIC instrumentation channel is considered to include the sensors and measurement channels for each Function, the operational bypass switches, and permissives. Failures that disable the capability to place a channel in operational bypass, but which do not disable the trip Function, do not render the protection channel inoperable.

The Bases for the LCO requirements of each specific EFIC Function are discussed next.

Loss of MFW Pumps

Four EFIC channels shall be OPERABLE with MFW pump turbines A and B control oil low pressure actuation setpoints of > 55 psig. The 55 psig setpoint is about half of the normal operating control oil pressure. The 55 psig setpoint Allowable Value appears to gave been arbitrarily chosen as a good indication of the Loss of MFW Pumps. Analysis Only assumes Loss of MFW Pumps and a specific value of MFW pump control oil pressure is not used in the analysis. Further, since the setpoint is so much less than

(continued)

BASES

LCO

Loss of MFW Pumps (continued)

operating control oil pressure, instrument error is not a consideration. The Loss of MFW Pumps Function includes a bypass enable and removal function utilizing the same bistable and auxiliary relay used in the NI/RPS bypass reactor trip on loss of both MFW pumps. However, the EFIC bypass is a logic requiring neutron flux to be < 20% RTP and the RPS to be in shutdown bypass. Practically speaking, the status of the bypass is strictly a function of the RPS shutdown bypass (i.e., required to be OPERABLE down into MODE 3).

OTSG Level -Low

Four EFIC dedicated low range level transmitters per OTSG shall be OPERABLE with OTSG Level- Low actuation setpoints of ≥ 0 inches indicated (nominally 8 inches above the top of the bottom tube sheet), to generate the signals used for detection for low level conditions for EFW Initiation. There is one transmitter for each of the four channels A, B, C, and D. The signals are also used after EFW is actuated to control at the low level setpoint of a nominal 32 inches above the lower tubesheet when one or more RCPs are in operation. In the determination of the low level setpoint, it is desired to place the setpoint as low as possible, considering instrument errors, to give the maximum operating margin between the ICS low load control setpoint and the EFW initiation setpoint. This minimizes spurious or unwanted initiation of EFW. To meet this criteria, a nominal setpoint of 8 inches indicated was selected, adjusted for potential instrument error, and shown to be conservative to the specified Allowable Value. Credit is only taken for low level actuation for those transients which do not involve a degraded environment. Therefore, normal environment errors only are used for determining the OTSG Level-Low Allowable Value.

OTSG Pressure-Low

Four OTSG Pressure-Low EFIC channels per OTSG shall be OPERABLE with an allowable value of ≥ 600 psig. The actual plant-setpoint is set higher to account for instrument loop uncertainties and calibration tolerances. The setpoint is chosen to avoid actuation under

(continued)

B 3.3 INSTRUMENTATION

B 3.3.13 Emergency Feedwater Initiation and Control (EFIC) Automatic Actuation Logic

BASES

BACKGROUND

Main Steam Line and Main Feedwater (MFW) Isolation

The four Emergency Feedwater Initiation and Control (EFIC) channels monitoring each Once Through Steam Generator (OTSG) outlet pressure condition input initiate commands to the actuation channels. FSAR Figure 7-26, (Ref. 1) illustrates the Main Steam Line and MFW Isolation Logics. The trip logic modules are physically located in the "A" and "B" EFIC channel cabinets. Channel "A" actuation logic initiates when instrumentation channel "A" or "B" initiates and channel "C" or "D" initiates, which in simplified logic is:

"A" actuation = (A and C) or (A and D) or (B and C) or (B and D).

Channel "B" actuation logic initiates when instrumentation channel "A" or "C" initiates and channel "B" or "D" initiates, which in simplified logic is:

"B" actuation = (A and B) or (A and D) or (C and B) or (C and D)

Each of the four Functions (OTSG A Main Feedwater Isolation, OTSG B Main Feedwater Isolation, OTSG A Main Steam Line Isolation, and OTSG B Main Steam Line Isolation) has a channel "A" and a channel "B" of automatic actuation logic.

Both channels "A" and "B" of the OTSG A Main Feedwater Isolation automatic actuation logic send closure signals to the OTSG A main feedwater pump suction valve, the three OTSG A block valves, and the MFW pump discharge cross connect valve. In addition, the instrumentation trips MFW pump "A."

Both channels "A" and "B" of the OTSG A Main Steam Line Isolation automatic actuation logic send closure signals to both of the OTSG A Main Steam Isolation valves.

(continued)

BASES

BACKGROUND

Main Steam Line and Main Feedwater (MFW) Isolation (continued)

OTSG B MFW and Main Steam Line Isolation automatic actuation logics respond similarly for the OTSG B valves and MFW pump B.

Emergency Feedwater (EFW) Actuation

The four EFIC instrumentation channels for each of the parameters being sensed input their initiate commands to the trip logic modules. FSAR Figure 7-26 Ref. 1) illustrates the EFW initiation logic. These trip logic modules are physically located in the "A" and "B" EF C channel cabinets.

EFW Actuation functions are the same logic combinations as MFW and Main Steam Line Isolation. Although not part of this Specification, EFW initiation also occurs on high pressure injection (HPI) actuation. Both channels of HPI actuation are input into each EFW actuation trip logic channel.

Vector Valve Enable Logic

The EFW module logic is responsible for sending open or close signals to the EFW control and block valves. FSAR Figure 7-26, Ref.1) illustrates the vector valve logic. The vector valve logic outputs are in a neutral state (neither commanding open nor close) until a signal is received from the vector valve enable Logic. The vector valve enable logic monitors the channel A and B EFW Actuation logics. When an EFW Actuation occurs, the vector enable logic enables the vector valve logic to generate open or close signals to the EFW valves depending on the relative values of OTSG pressures.

APPLICABLE SAFETY ANALYSES

Automatic isolation of MFW and main steam line is assumed in the safety analyses to mitigate the consequences of main steam line or MFW line breaks. The SLB analysis was re-performed crediting all "as built" systems and their associated response times. This is documented in AREVA Engineering Information Record 51-9108715-000, "CR-3 ROTSG Support - Limited Scope DOE." In this analysis, EFIC isolation of Main Feedwater and Main Steam were credited. The worst case evaluated single failure of the feedwater isolation system was determined to be a failure of the MFP to trip. This accident is terminated by the closing of the MFP suction valves and the downstream block valves are not credited

(continued)

BASES

BACKGROUND (continued)

The valve open/close commands are determined by the relative values of OTSG pressures as follows:

PRESSURE STATUS	VECTOR VALVES "A"	VECTOR VALVES "B"
If OTSG "A" & OTSG "B" > 600 psig	Open	Open
If OTSG "A" > 600 psig & OTSG "B" < 600 psig	Open	Close
If OTSG "A" < 600 psig & OTSG "B" > 600 psig	Close	Open
If OTSG "A" & OTSG "B" < 600 psig		
AND		
OTSG "A" & OTSG "B" within 125 psid	Open	Open
OTSG "A" 125 psid > OTSG "B"	Open	Close
OTSG "B" 125 psid > OTSG "A"	Close	Open

APPLICABLE SAFETY ANALYSES

The SLB analysis was re-performed crediting all "as built" systems and their associated response times. This is documented in AREVA Engineering Information Record 51-9108715-000, “CR-3 ROTSG Support - Limited Scope DOE.” In this analysis, EFIC isolation of Main Feedwater and Main Steam were credited.The Vector Valve Logic, Feed Only Good Generator (FOGG) was credited for termination of an overfeed condition for either OTSG during postulated DC power failures. No operator action was credited or required.

EFW vector valve logic response time is included in the response time for each EFW instrumentation Function and is not specified separately.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The EFIC-EFW-vector valve logic satisfies Criterion 3 of the NRC Policy Statement.

LCO

Four channels of the EFIC-EFW-vector valve logic are required to be OPERABLE in order to provide the dual function of the valves while meeting single failure criteria. Refer to the ACTIONS for further discussion of the two functions. The 600 psig and 125 psid Allowable Values were chosen as discussed in the Bases for Specification 3.3.11, "EFIC System Instrumentation." The feed only good generator verification study assumed a differential pressure vector value of 150 psid. The 125 psid setpoint conservatively assumes a 25 psi margin for instrument error. Failure to meet this LCO results in not being able to meet the single-failure criterion.

APPLICABILITY

EFIC-EFW-vector valve logic is required in MODES 1, 2, and 3 because the OTSGs are relied on in these MODES for RCS heat removal. In MODES 4, 5, and 6, heat removal requirements are reduced and may be provided by the Decay Heat Removal System. Therefore, vector valve logic is not required to be OPERABLE in these MODES.

ACTIONS

A.1

The function of the EFIC-EFW control/block valves and the vector valve logic is to meet the single-failure criterion while maintaining the capability to:

a. Provide EFW to an intact OTSG on demand; and

b. Isolate a faulted OTSG when required.

These conflicting requirements result in the necessity for two valves in series, in parallel with two valves in series, and a four channel valve command system.

(continued)

BASES

LCO

14,15. Steam Generator Water Level (Start-up Range and Operating Range

The CR-3 Type A/Category 1 indication of steam generator level is the startup range and operating range EFIC level instrumentation. The combined instrument ranges cover a span of a nominal 8 to 396 inches above the lower tubesheet. The measured low range differential pressure is displayed in inches of water. The low range indicates a range of 0 to 150 inches, where 0 inches indicates an actual nominal level of 8 inches above the lower tubesheet. The high range steam generator level instrumentation indicates a span of 0 to 100%, where 0% corresponds to a nominal 104 inch actual level above the lower tubesheet. Redundant monitoring capability is provided by two channels of each range of instrumentation per OTSG.

The level signals are displayed on control room indicators. The steam generator level signals are calculated from differential pressure signals which are pressure compensated by a module in the EFIC System cabinets. Compensation is based on the densities of the water and steam assuming the OTSGs are normally operating at saturation. Each operating range level transmitter also inputs to a recorder in the control room. Since operator action is based on the control room indication, the LCO deals specifically with this portion of the instrument string.

(continued)

BASES

LCO

16. Steam Generator Pressure

Steam generator pressure is measured on each main steam line between the respective main steam safety valves and the main steam isolation valve. Redundant monitoring capability is provided by two pressure transmitters per OTSG. Each pressure transmitter provides an input signal to pressure indicators and a recorder in the control room. The control room indication of OTSG pressure is one of the primary indications used by the operator during an accident. Therefore, the LCO deals specifically with the control room indication portion of the OTSG pressure instrument string. The range of the indication is 0 to 1200 psig.

OTSG pressure decreases rapidly during a design basis steam line break accident. This rapid decrease in pressure is a positive indication of a breach in the secondary system pressure boundary. In order to minimize the primary system cooldown caused by the decreasing secondary system pressure, feedwater flow to the affected OTSG must be terminated. OTSG pressure is considered a Type A variable because it is the primary indication used by the operator to identify and isolate the affected OTSG. In addition, OTSG pressure is a key parameter used by the operator to evaluate primary-to-secondary heat transfer. For example, the operator may use this indication to control the primary system cooldown following a steam generator tube rupture or a small break loss of coolant accident (LOCA).

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 Steam Generator (OTSG) Tube Integrity

BASES

BACKGROUND

Steam generator (OTSG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The OTSG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The OTSG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the OTSG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the OTSG. The OTSG heat removal function is addressed by LCO 3.4.4, "RCS Loops - MODE 3," LCO 3.4.5, "RCS Loops - MODE 4," LCO 3.4.6, "RCS Loops - MODE 5, Loops Filled," and is implicitly required in MODES 1 and 2 in order to prevent a Reactor Protection System actuation (LCO 3.3.1).

OTSG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The OTSG performance criteria are used to manage OTSG tube degradation.

Specification 5.6.2.10, "Steam Generator (OTSG) Program," requires that a program be established and implemented to ensure that OTSG tube integrity is maintained. Pursuant to Specification 5.6.2.10, tube integrity is maintained when the OTSG performance criteria are met. There are three OTSG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The OTSG

(continued)

BASES

BACKGROUND (continued)	performance criteria are described in Specification 5.6.2.10. Meeting the OTSG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions. The processes used to meet the OTSG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).
APPLICABLE SAFETY ANALYSES	The steam generator tube rupture (SGTR) accident is the limiting design basis event for OTSG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.12, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser. The analysis for design basis accidents and transients other than a SGTR assume the OTSG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all OTSGs of one gallon per minute or is assumed to increase to one gallon per minute as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.15, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 50.67 (Ref. 3) or the NRC approved licensing bases (e.g., a small fraction of these limits). Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The LCO requires that OTSG tube integrity be maintained. The LCO also requires that all OTSG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

(continued)

BASES

LCO (continued)

During an OTSG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, an OTSG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

An OTSG tube has tube integrity when it satisfies the OTSG performance criteria. The OTSG performance criteria are defined in Specification 5.6.2.10, "Steam Generator Program," and describe acceptable OTSG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the OTSG performance criteria.

There are three OTSG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the OTSG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure

(continued)

BASES

LCO (continued)

is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed one gallon per minute per OTSG. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of OTSG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.12, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one OTSG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

(continued)

BASES

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across OTSG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each OTSG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected OTSG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected OTSG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more OTSG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.16.2. An evaluation of OTSG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the OTSG performance criteria described in the Steam Generator Program. The OTSG repair criteria define limits on OTSG tube degradation that allow for flaw growth between inspections while still providing assurance that the OTSG performance criteria will continue to be met. In order to determine if an OTSG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the OTSG performance criteria will continue to be met until the next refueling outage or OTSG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next OTSG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with an OTSG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or OTSG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or OTSG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if OTSG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

During shutdown periods the OTSGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During OTSG inspections a condition monitoring assessment of the OTSG tubes is performed. The condition monitoring assessment determines the “as found” condition of the OTSG tubes. The purpose of the condition monitoring assessment is to ensure that the OTSG performance criteria have been met for the previous operating period.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1 (continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the OTSG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.16.1. The Frequency is determined by the operational assessment and other limits in the OTSG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the OTSG performance criteria at the next scheduled inspection. In addition, Specification 5.6.2.10 contains prescriptive requirements concerning inspection intervals to provide added assurance that the OTSG performance criteria will be met between scheduled inspections.

SR 3.4.16.2

During an OTSG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 5.6.2.10 are intended to ensure that tubes accepted for continued service satisfy the OTSG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the OTSG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the OTSG performance criteria.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.16.2 (continued)

The Frequency of prior to entering MODE 4 following a OTSG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the OTSG tubes to significant primary to secondary pressure differential.

REFERENCES

1. NEI 97-06, “Steam Generator Program Guidelines.”
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 50.67.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Draft Regulatory Guide 1.121, “Basis for Plugging Degraded Steam Generator Tubes,” August 1976.
6. EPRI, “Pressurized Water Reactor Steam Generator Examination Guidelines.”

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1 (continued)

The 31 day frequency is appropriate because the valves are operated under administrative control. This frequency has been shown to be acceptable through operating experience.

SR 3.5.2.2

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME OM Code (Ref. 4). This type of testing may be accomplished by measuring the pump's developed head at only one point of the pump's characteristic curve and this point may be anywhere on the curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant accident analysis. SRs are specified in the Inservice Testing Program, which encompasses the ASME OM Code. The ASME OM Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.3 and SR 3.5.2.4

These SRs demonstrate that each automatic ECCS valve that is not locked, sealed, or otherwise secured in position, actuates to its required position on an actual or simulated ESAS signal and that each ECCS pump starts on receipt of an actual or simulated ESAS 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. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the ESAS testing, and equipment performance is monitored as part of the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.5

Verification of the positions of the listed valves in the HPI flowpath ensures adequate flow resistance in the overall system and the individual HPI lines. Maintenance of adequate flow resistance and pressure drop in the piping system for each injection point is necessary in order to: (1) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS LOCA analyses; (2) provide an acceptable level of total ECCS flow to all injection points equal to or above values assumed in the ECCS LOCA analyses; (3) ensure adequate cooling flow to the HPI pump mechanical seals; and (4) prevent HPI pump flow from exceeding 600 gpm when the system is in its minimum resistance configuration (600 gpm is the maximum HPI pump flow rate assumed in design calculations associated with Emergency Diesel Generator loading, ECCS pump available NPSH, and makeup tank (MUT-1) allowable overpressure versus level). This 24 month Frequency is acceptable based on consideration of the design reliability of valves that are locked, sealed, or otherwise secured in position.

Verification of correct valve position will be accomplished by assuring the mechanism that locks, seals or secures the valves is intact. If the stop check valves or throttle valves are repositioned, the valves must be returned to their correct position and then secured. This "as-left" position verification ensures the HPI flow assumptions in the accident analysis are maintained.

SR 3.5.2.6

This Surveillance ensures that the flow controllers for the LPI throttle valves will automatically control the LPI train flow rate in the desired range and prevent LPI pump runout as RCS pressure decreases after a LOCA. The 24 month Frequency is acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

(continued)

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 2619 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.

(continued)

BASES

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

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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 AREVA Engineering Information Record 51-9108715-000, “CR-3 ROTSG Support - Limited Scope DOE.” 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 595 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)

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs)

BASES

BACKGROUND

The main feedwater isolation valves (MFIVs) are designated valves in the Main Feedwater (MFW) System which function in conjunction with other equipment to isolate MFW to the steam generators (OTSGs) in accordance with assumptions used in the high energy line break accident analyses.

At CR-3, the MFIVs for each OTSG consist of the MFW pump suction valve, the main/startup/low load block valves (in parallel), and the MFW pump discharge cross connect valve between OTSG A and B (Ref. 1). All the OTSG A valves receive a signal to close on low OTSG pressure from EFIC OTSG A MFW isolation automatic actuation logic channels A and B. OTSG B valves similarly receive signals from EFIC OTSG B MFW isolation automatic actuation logic channels A and B. The crossover valve receives closure signals from both channels of EFIC's OTSG A and OTSG B MFW isolation logics (Ref. 2).

In addition to the above, the EFIC OTSG A MFW isolation logic trips MFW pump A on a OTSG A low pressure signal. OTSG B EFIC logic trips MFW pump B on OTSG B low pressure. EFIC also provides a trip of the opposite side MFW pump and closure of its suction valve on a single side feedwater isolation signal when the crossover valve is open. This logic is enabled by manual key switches which are administratively controlled during times when both OTSGs are being fed from one MFW pump (typically below 55% power). This reduces the system pressure on the MFW pump startup, crosstie MFW pump discharge valve, and low load block valves and assures these MFIVs can perform their intended function. The tripping of the other train assures main feedwater isolation for the case where the EFIC initiation is for the opposite side from the operating MFW pump.

This results in several layers of redundancy in that not only are the fluid system components (valves, pumps) redundant, but the automatic closure signals to each component are also redundant.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Closure of the MFIVs terminates the addition of feedwater to an affected OTSG. This limits the mass and energy releases for breaks within containment, reduces cooldown effects, and reduces the potential for a return to power due to a return to critical following reactor trip.

The SLB analysis was reperformed crediting all "as built" systems and their associated response times. This is documented in AREVA Engineering Information Record 51-9108715-000, "CR-3 ROTSG Support - Limited Scope DOE." In this analysis, EFIC isolation of Main Feedwater and Main Steam were credited. The required stroke time of the MFIVs, except for the low load block valves FWV-31 and FWV-32, is 34 seconds which includes an EFIC signal process delay and valve closure from the time of OTSG low pressure of 595 psig. The actual EFIC actuation on OTSG low pressure is greater than or equal to 600 psig. The lower analysis pressure is conservative. The low load block valves FWV-31 and FWV-32 are required to stroke close in 67 seconds which includes an EFIC signal process delay and valve closure.

The MFIVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO ensures that the MFIVs will isolate MFW flow to the OTSGs following a FWLB or a main steam line break. The following valves are addressed by this LCO:

<u>OTSG A</u>		<u>OTSG B</u>
FWV-30	Main block valve	FWV-29
FWV-31	Low load block valve	FWV-32
FWV-36	Startup block valve	FWV-33
FWV-14	MFW pump suction valve	FWV-15

FWV-28 MFW cross connect valve

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

With one or more flow paths not capable of isolating within the required isolation time, action must be taken to restore one valve to within the required closure time or to isolate the affected flow path. The Required Actions are the same as those specified in Condition A of this Specification, except the Completion Time for B.1 is reduced to 24 hours.

The 24 hour Completion Time reflects analysis that demonstrated reduced stroke times will not likely challenge the containment analysis. However, the level of degradation represented by this Condition is considered more serious than Condition A.

When in Condition B, the Required Actions of Condition A are also applicable.

C.1

With two inoperable valves in the same flow path, valve isolation capability has been lost. Under these conditions, at least one of the affected valves in each flow path must be restored to OPERABLE status within 8 hours. The 8 hour Completion Time is reasonable, based on operating experience.

ACTIONS (continued)

D.1

With a startup block valve (FWV-33,-36) in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status within 72 hours. In this Condition, valve closure or isolation is not an acceptable alternative action because the ICS controlled startup control valves are necessary to provide and control main feedwater following a reactor trip. Closure of the startup block valves would preclude this function post-trip and potentially challenge Emergency Feedwater System operation.

(continued)

BASES

ACTIONS

D.1 (continued)

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valve in the flowpath, the MFW ump trip feature provided on low OTSG pressure, and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

E.1 and E.2

If the MFIVs cannot be restored to OPERABLE status, or closed, or isolated 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 3.7.3.1

This SR verifies that MFIV closure time is within the acceptance criteria in the Inservice Testing Program. In order to be consistent with the safety analysis as documented in AREVA Engineering Information Record 51-9108715-000, "CR-3 ROTSG Support - Limited Scope DOE." the required stroke time of the MFIVs, except for the low load block valves FWV-31 and FWV-32, is 34 seconds which includes an EFIC signal process delay and valve closure from the time of OTSG low pressure of 595 psig. The actual EFIC actuation on OTSG low pressure is greater than or equal to 600 psig. The lower analysis pressure is conservative. The low load block valves FWV-31 and FWV-32 are required to stroke close in 67 seconds which includes an EFIC signal process delay and valve closure. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The MFIVs should 1 not be tested at power since even a part stroke exercise increases the risk of a valve closure, and the risk of a plant transient with the plant generating power. As these valves are not tested at power, they are exempt from the ASME OM Code (Ref. 3) quarterly valve stroke requirements.

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 the test conditions most representative of those under which the acceptance criterion was generated.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.17 Steam Generator Level

BASES

BACKGROUND

The principal operational function of the steam generators (OTSGs) is to provide superheated steam at a constant pressure (900 psia) over the power range. OTSG water inventory is maintained large enough to provide adequate primary to secondary heat transfer. Mass inventory and indicated water level in the OTSG increases with load as the length of the four heat transfer regions within the OTSG vary. Inventory is controlled indirectly as a function of power and maintenance of a constant average primary system temperature by the feedwater controls in the Integrated Control System.

The maximum operating OTSG level is based primarily on preserving the initial condition assumptions for OTSG inventory used in the FSAR steam line break (SLB) analysis (Ref. 1). An inventory of 62,600 lbm was used in this initial analysis for the original OTSG. The 62,600 lbm was based upon concerns of a possible return to criticality because of primary side cooling following an SLB and the maximum pressure in the reactor building. Subsequently, the SLB was re-analyzed to reflect updated information which indicated it was not possible to put 62,600 lbm in the OTSG without putting water in the steam lines. A value of 56,340 lbm was used in the reanalysis of the previously listed concerns.

For a clean original OTSG, the mass inventory in the OTSG operating at 100% power is approximately 39,000 lbm to 40,000 lbm. For a clean replacement OTSG, the mass inventory at 100% power is approximately 48,000 lbm. As an OTSG becomes fouled and the operating level approaches the limit of 96%, the mass inventory in the downcomer region increases approximately 7,000 lbm (Ref. 2), and adds to the total mass inventory of the OTSG. In matching unit data of startup level versus power, OTSG performance codes have shown that fouling of the lower tube support plates does not significantly change the heat transfer characteristics of generator. Thus, the steam temperature, or superheat, is not degraded due to the fouling of the tube support plates, and mass inventory changes are mainly due to the added level in the downcomer.

Analytically, increasing the fouling of the OTSG tube surfaces degrades the heat transfer capability, increases the mass inventory, and decreases the steam superheat at

(continued)

BASES

BACKGROUND (continued)

100% power. The results were presented as the amount of mass inventory in each steam generator versus operating range level and steam superheat.

The limiting curve, which was determined from several steam generator performance code runs at a power level of 100%, conservatively bounds steam generator mass inventory value when operating at power levels < 100%.

The points displayed in Figure 3.7.17-1, in the accompanying LCO, are the intercept points of the 56,340 lb mass value and the operating range level and steam superheat values for the original OTSGs. For the replacement OTSGs a secondary side inventory of 62,000 lbm is bounded by Figure 3.7.17-1. The updated SLB analysis using a secondary side inventory ≥ 62,000 lbm in the replacement OTSG shows all acceptance criteria for the event were met (Reference 3).

The OTSG performance analysis also indicated that startup and full range level instruments are inadequate indicators of steam generator mass inventory at high power levels due to the combination of static and dynamic pressure losses. If the water level should rise above the 96% upper limit, the steam superheat would tend to decrease due to reduced feedwater heating through the aspirator ports. Normally, a reduction in water level is manually initiated to maintain steam flow through the aspirator port by reducing the power level. Thus, the superheat versus level limitation also tends to ensure that, in normal operation, water level will remain clear of the aspirator ports.

Feedwater nozzle flooding would impair feedwater heating, and could result in excessive tube to shell temperature differentials, excessive tubesheet temperature differentials, and large variations in pressurizer level.

APPLICABLE SAFETY ANALYSES

The limiting Design Basis Accident with respect to OTSG operating level is a steam line break (Reference 1). The parameter of interest is the mass of water, or inventory, contained in the steam generator due to its role in lowering Reactor Coolant System (RCS) temperature (return to criticality concern), and in raising containment pressure during an SLB accident. A larger inventory causes the effects of the accident to be more severe. Figure 3.7.17-1, in the accompanying LCO, was evaluated for the replacement OTSGs based upon maintaining inventory < 62,000 lbm. The replacement OTSG inventory was evaluated for SLB as described in Reference 3 and found to be acceptable when compared to the acceptance criteria of Reference 1.

(continued)

BASES (continued)

LCO

This LCO preserves the initial condition assumptions of the accident analyses. Failure to meet the maximum OTSG level LCO requirements can result in additional mass and energy released to containment, and excessive cooling (and related core reactivity effects) following an SLB.

APPLICABILITY

In MODES 1 and 2, a maximum OTSG water level is required to preserve the initial condition assumption for OTSG inventory used in the steam line failure accident analysis (Ref. 1).

In MODES 3, 4, and 5, limits on SHUTDOWN MARGIN (Refer to LCO 3.1.1) and the corresponding RCS boron concentration considers OTSG level and prevents a return to criticality in the event of an SLB. In MODE 6, the water in the OTSG has a low specific enthalpy/energy; therefore, there is no need to limit the OTSG inventory when the plant is in this condition.

ACTIONS

A.1

With OTSG level in excess of the maximum limit, action must be taken to restore the level to within the bounds assumed in the safety analysis. The 15 minute Completion Time was established considering that while inadequate SDM may exist in this condition, the probability of an SLB occurring during this time frame is low. 15 minutes is also considered adequate time to reduce THERMAL POWER and restore compliance with the limit.

B.1

If the water level in one or more steam generators cannot be restored to less than or equal to the maximum level in Figure 3.7.17-1, the plant must be placed in a MODE that minimizes the accident risk. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours. The Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS

B.1 (continued)

It is likely that as power is reduced, OTSG level will be restored to within the limit of Figure 3.7.17-1. When this occurs and level is restored to within the limit, the power reduction may be terminated in accordance with LCO 3.0.2.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies OTSG level to be within acceptable limits. The 12 hour Frequency is adequate considering levels vary very little while operating at steady-state conditions. During non-steady state conditions, the operator would likely be aware of any significant variations in OTSG level. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to OTSG level status.

REFERENCES

1. FSAR, Section 14.2.2.1.
2. Calculation M09-0012, ROTSG Thermal-Hydraulic Performance for Original Full Power and MUR Conditions
3. Calculation M09-0019, ROTSG Disposition of Events, Attachment 1

BASES

LCO (continued)

inventory supports the availability of the DD-EFW Pump to fulfill its mission of supplying EFW flow to one or both steam generators. The DD-EFW pump is required to provide emergency feedwater to one or two steam generators under the EFIC flow control scheme for an anticipated operational occurrence (AOO) or a postulated DBA with loss of offsite power.

The starting air system is required to have a minimum capacity for six successive engine 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

Emergency feedwater flow is required during a Small Break LOCA or loss of main feedwater in order to cool and depressurize one or both generators which supports the reactor shut down and maintains 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 DD-EFW Pump OPERABILITY, these features are required to be within limits whenever the DD-EFW pump is required to be OPERABLE.

ACTIONS

A.1

With total fuel oil volume in the supply tank < 9,580 gallons and > 8,435 gallons, there is enough fuel oil available to operate the DD-EFW pump for 6 days. 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 DD-EFW Pump inoperable.

The 48 hour Completion Time allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. This period is acceptable based on the remaining capacity (> 6 days), the actions that will be initiated to obtain replenishment, and the low probability of an event occurring during this brief period.

(continued)

BASES

ACTIONS

B.1

With stored lube oil inventory between 178 and 207 gallons, there is not sufficient lube oil to support 7 days continuous operation of the DD-EFW Pump. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6 day supply. In this Condition, a period of 48 hours is considered adequate to restore the required volume prior to declaring the DD-EFW Pump inoperable. The volume of stored lube oil specified does not include the engine lube oil inventory contained in the sump. If the required stored volume cannot be restored, the DD-EFW Pump is declared inoperable.

The 48 hour Completion Time is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the actions that will be initiated to obtain replenishment, and the low probability of an event occurring during this brief period.

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion for DD-EFW Pump fuel oil particulates. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. However, poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean the fuel oil will not burn properly and given that proper engine performance has been recently demonstrated (per SR 3.7.5.2), it is prudent to allow a brief period of time prior to declaring the associated DD-EFW Pump inoperable. The 7 day Completion Time allows for further evaluation, resampling, and re-analysis of the DD-EFW Pump fuel oil.

(continued)

BASES

BACKGROUND (continued)

Provided an ES signal is present, certain required ES loads are returned to service in a predetermined sequence in order to prevent overloading the EDG in the process. Within 35 seconds after the initiating signal is received, all loads needed to recover the plant or maintain it in a safe condition are returned to service.

The service ratings of the EDG are:

0 to 2850 kw on a continuous basis

2851 to 3200 kw on a cumulative 2000 hour basis

3201 to 3400 kw on a cumulative 200 hour basis

3401 to 3500 kw on a cumulative 30 minute basis.

Loads powered from the 4160 V ES buses are listed in Reference 2.

Steady state load does not include loads imposed by the starting of motors such as during block loading, and short duration loads such as motor operated valves, battery charger surges, and short duration pump surge flows. The EDG engines have a mechanical power capability of 3910 KW at 900 rpm for motor starting.

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the FSAR, Chapter 6 (Ref. 4) and Chapter 14 (Ref. 5), assume ES systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ES systems so that the fuel, RCS, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and the design basis of the plant. This results in maintaining at least one train of the onsite or offsite AC sources OPERABLE during accident conditions in the event of:

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

(continued)

BASES

ACTIONS

E.1

With the Train A and Train B EDGs inoperable, there are no qualified onsite standby AC sources. Thus, with an assumed loss of offsite electrical power, there would not be sufficient standby AC sources available to power the minimum required ES systems. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time is balanced with that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). However, since any inadvertent generator trip could also result in a total loss of offsite AC power, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

The 2 hour Completion Time is consistent with the recommendations of Reference 6.

F.1 and F.2

If the inoperable AC electrical power sources 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 12 hours and in MODE 5 within 36 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.

G.1

Condition G corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any subsequent failures in the AC electrical power system will cause a loss of function condition, and potentially, a station blackout. Therefore, the unit is required to enter LCO 3.0.3 immediately and prepare for a controlled shutdown.

(continued)

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function. This is consistent with 10 CFR 50, Appendix A, GDC 18 (Ref. 8). Periodic component tests are supplemented by extensive functional tests during outages (under simulated accident conditions). Where the SRs for this LCO specify voltage and frequency tolerances, the following is applicable. The specified steady state minimum and maximum voltage (4077 V - 4243 V) and frequency (59.4 Hz - 60.6 Hz) supports ES loads performing their design function and supports acceptable EDG loading. The specified voltage and frequency also support dynamic voltage and frequency dips resulting from motor starting being no greater than 25% and 5% respectively, and voltage and frequency recovering within 3 seconds to within 10% and 2% respectively. The steady state voltage and frequency values are derived from plant specific calculations. The voltage and frequency dip percentages and the recovery times and transient recovery percentages are derived from the recommendations in Regulatory Guide 1.9, (Ref. 3).

The limits used to define when the EDG is “started” for the ten second EDG start requirement will be 90% voltage (3744 V) and 98% frequency (58.8 Hz). These values are based on Regulatory Guide 1.9, Revision 3, July 1993, Section C.1.4 which addresses voltage and frequency recovery during EDG block loading prior to the next load block being applied to the EDG. The CR-3 ready matrix relays are set at 90.8% voltage and 98% frequency.

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected or are capable of being connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

SR 3.8.1.2 and SR 3.8.1.6

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the plant in a safe shutdown condition. To minimize wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note (Note 2 for SR 3.8.1.2) to indicate that all EDG starts for these Surveillances may be preceded by an engine prelube period and followed by a warmup period prior to loading.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

<u>SR 3.8.1.2 and SR 3.8.1.6</u> (continued)

For the purposes of SR 3.8.1.2 and SR 3.8.1.6 testing, the EDGs are started from standby conditions. Standby conditions for an EDG means that the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with the manufacturer's recommendations.

SR 3.8.1.6 requires that, at a 184 day Frequency, the EDG starts from standby conditions and achieves required voltage and frequency within 10 seconds. The 10 second start requirement supports the assumptions of the design basis LOCA analysis in the FSAR, Chapter 14 (Ref. 5).

The 10 second start requirement is not applicable to SR 3.8.1.2 (see Note 3) when a modified start procedure is used. If a modified start is not used, the 10 second start requirement of SR 3.8.1.6 applies.

The monthly modified (slow) start of the EDG is to demonstrate that the EDG will run and achieve the required voltage and frequency (no start time requirement). A note that allows control of the speed (frequency) using automatic, manual, or a combination of the two mechanisms differentiates the slow start from the fast start, where manual adjustment of the speed setting for the governor to meet the test requirement frequency would not be appropriate. The fast start test assures the governor speed as-left setting from the last test and the governor time response performance are acceptable. The governor speed as-left setting is procedurally controlled regardless of whether the previous test was a fast start or a slow start. Control of the EDG frequency for the slow start can be from the EDG Speed Switch, the EDG Start Mode Select Switch, or the Speed Setting Knob located on the governor.

In addition to the SR requirements, the time for the EDG to reach steady state operation, unless the modified EDG start method is employed, is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

Since SR 3.8.1.6 requires a 10 second start, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2. This is the intent of Note 1 of SR 3.8.1.2.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.2 and SR 3.8.1.6 (continued)

The 31 day Frequency for SR 3.8.1.2 is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3). The 184 day Frequency for SR 3.8.1.6 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 7). These Frequencies provide adequate assurance of EDG OPERABILITY, while minimizing testing-induced degradation.

SR 3.8.1.3

This Surveillance verifies that the EDGs are capable of providing power near the upper end of the continuous rating of the machines. The surveillance is performed by synchronizing and paralleling the EDGs with offsite power sources to facilitate loading the EDGs. The EDG synch check relays, which supervise closing the EDG breakers during surveillance testing, are not required for EDG OPERABILITY since they are not required for the EDGs to perform their design function. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the EDG is connected to the offsite source.

Although no power factor requirements are established by this SR, the EDG is normally operated at a slightly positive power factor. The power factor (reflected in reactive loading) is an operational limitation to provide optimum control over the EDGs.

The 31 day Frequency for this Surveillance is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3).

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients outside the load range (e.g., due to changing bus loads) do not invalidate this test. Note 3 indicates that this Surveillance only be conducted on one EDG at a time 1n order to minimize the potential for common cause failures that occur as a result of offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful EDG start must precede this test to credit satisfactory performance.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.10 (continued)

This SR is modified by three Notes. The reason for Note 1 is to minimize wear and tear on the EDGs during testing. For the purpose of this testing, the EDGs may be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for EDGs. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and potentially challenge safety systems. However, Note 2 acknowledges that should an unplanned event occur in MODES 1, 2 or 3, following verification that the acceptance criteria of the SR are met, the event can be credited as a successful performance of this SR. Note 3 is an SR 3.0.4 type allowance to place the plant in MODE 4 for the purposes of performing this Surveillance. This is necessary in order to establish the pre-requisite plant configuration needed to perform the SR.

SR 3.8.1.11

This Surveillance demonstrates the EDGs are capable of providing power greater than or equal to the maximum expected steady state accident loads, which are the automatically connected accident loads and required manually applied accident loads. However, the upper limit of the 200 hour service rating is still available for flexibility in post accident EDG load management, including short duration loads. The test load band is provided to avoid routine overloading of the EDGs. Routine overloading may result in more frequent teardown inspections, in accordance with vendor recommendations, in order to maintain EDG OPERABILITY. The EDG synch check relays are not a required component for EDG OPERABILITY. They are utilized to facilitate parallel loading for testing and are not required for the EDG to perform its design function.

The 60 minute run time is provided to stabilize the engine temperature. This ensures that cooling and lubrication are adequate for extended periods of operation.

The 24 month Frequency takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with expected fuel cycle lengths.

This Surveillance is modified by two Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. The reason for Note 2 is that during

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, plant safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following periodic governor replacement, corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment. However, the Note acknowledges that credit may be taken for unplanned events that satisfy this SR.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. FSAR, Chapter 8.
3. Regulatory Guide 1.9, Rev. 3, July 1993.
4. FSAR, Chapter 6.
5. FSAR, Chapter 14.
6. Regulatory Guide 1.93, Rev. 0, December 1974.
7. Generic Letter 84-15.
8. 10 CFR 50, Appendix A, GDC 18.
9. Regulatory Guide 1.108, Rev. 1, August 1977.
10. Regulatory Guide 1.137, Rev. 1, October 1979.
11. ANSI C84.1-1982.
12. ASME Code for the Operation and Maintenance of Nuclear Power Plants (ASME OM Code).
13. Deleted.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters - Operating

BASES

BACKGROUND

The inverters are the preferred source of power for the AC vital buses because of the stability and reliability they provide by being powered-from the 125 VDC battery source. The function of the inverter is to convert DC electrical power to AC electrical power, thus providing an uninterruptible power source (UPS). The 120 VAC vital bus distribution system is used to supply four separate channels of 120 volt instrument power to equipment such as the Nuclear Instrumentation (NJ) System, the Engineered Safeguards Actuation System (ESAS) and the Reactor Protection System (RPS). Other loads fed by 120 volt vital AC power include communication equipment, valves and relays essential to safe plant operation and shutdown.

Each of the four UPS bus sections are supplied through a dual input inverter (with internal auctioneering circuit), or a 480 V ES motor control center (MCC) through a transformer. The dual input static inverters 3A, 3B, 3C, and 3D are normally supplied from a 480 V ES MCC with an uninterrupted transfer to a 125 V battery source on loss of normal supply. Static inverters B and D are normally fed from MCC 3B2 or alternately fed from the B battery train. Static inverters A and C are normally fed from MCC 3A1 or alternately from the A battery train. The power supplied through the transformers to the UPS buses comes from MCCs 3A2 or 3B1.

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 1), and Chapter 14 (Ref. 2), assume Engineered Safeguard systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the design basis of the plant. This includes maintaining required AC vital buses OPERABLE during accident conditions in the event of:

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

b. A worst-case single failure.

Inverters are a part of the electrical power distribution system. As such, they satisfy Criterion 3 of the NRC Policy Statement.

LCO

The inverters ensure the availability of AC electrical power for the components and instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESAS instrumentation and controls is maintained. The four battery powered inverters (two per train) ensure an uninterruptible supply of AC electrical power to the AC vital buses in the event the 4160 V ES buses are de-energized.

In order to consider inverters OPERABLE, the associated AC vital bus must be powered by the inverter and the correct DC voltage applied from a battery to the auctioneering circuit. The inverter, via the internal auctioneering circuit, is normally powered from the associated Class 1E 480 V ES bus. In this case, the auctioneering circuit must also be capable of switching to the required DC supply (i.e., the associated 125 VDC Class 1E battery). Additionally, inverter output AC voltage and frequency must be within established tolerances.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

During movement of fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident involving handling recently irradiated fuel. For Cycles (including the following refueling outage) operated at or below 2619 MWth (RATED THERMAL POWER plus heat balance uncertainty), recently irradiated fuel is the fuel that has occupied part of a critical reactor core within the previous 72 hours. 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," in conjunction with the administrative limit on minimum decay time prior to irradiated fuel movement ensure that the release fission product radioactivity subsequent to a fuel handling accident results in doses that are within the requirements specified in 10 CFR 50.67 even without containment closure.

Containment penetrations satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO limits the consequences of a fuel handling accident involving handling recently irradiated fuel 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, including the equipment hatch or the Outage Equipment Hatch, to be closed except for penetrations containing an OPERABLE purge or mini-purge valve. For the containment purge and mini-purge valves to be considered OPERABLE, at least one valve in each penetration must be automatically isolable on an RB Purge-high radiation isolation signal.

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.

APPLICABILITY

The containment penetration requirements are applicable during movement of recently irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment." In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is not being conducted; the potential for a fuel handling accident does not exist. Additionally, due to radioactive decay, a fuel handling accident involving fuel that has not been recently irradiated will result in doses that are well within the guideline values specified in 10 CFR 50.67 even without containment closure capability. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1

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 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 needed. This is accomplished by immediately suspending movement of recently irradiated fuel assemblies within containment. Performance of these actions shall not preclude moving a component to a safe position.

(continued)