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10 CFR 50.90

PG&E Letter DCL-15-152

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

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to NRC Request for Additional Information Regarding License Amendment Request 15-03, "Application of Alternative Source Term"

References:

1. PG&E Letter DCL-15-069, License Amendment Request 15-03, "Application of Alternative Source Term," dated June 17, 2015 (ADAMS Accession No. ML15176A539)
2. PG&E Letter DCL-15-105, "Supplement to License Amendment Request 15-03, 'Application of Alternative Source Term,'" dated August 31, 2015 (ADAMS Accession No. ML15243A363)
3. E-mail from NRC Project Manager Siva P. Lingam, "Diablo Canyon 1 and 2 - Requests for Additional Information for License Amendment Request 15-03 to Adopt the Alternative Source Term per 10 CFR 50.67 (TAC Nos. MF6399 and MF6400)," dated October 20, 2015

Dear Commissioners and Staff:

License Amendment Request (LAR) 15-03, "Application of Alternative Source Term," was submitted by Pacific Gas and Electric (PG&E) Letter DCL-15-069 (Reference 1) and supplemented by PG&E Letter DCL-15-105 (Reference 2).

In Reference 3, the NRC Reactor Systems Branch (SRXB) requested additional information required to complete the review of LAR 15-03. PG&E's responses to the SRXB staff's questions are provided in the Enclosure.

The Enclosure to this letter provides the following attachments:

- Attachment 1 – Technical Specification Bases Markup (For Information Only), Revision 1

A001
NRR

- Attachment 2 - Diablo Canyon Power Plant Technical Report Prepared by CB&I Stone and Webster, Inc., "Implementation of Alternative Source Terms, Summary of Dose Analyses and Results," Revision 1
- Attachment 3 – Regulatory Guide 1.183 Conformance Tables, Revision 1
- Attachment 4 – Diablo Canyon Power Plant Updated Final Safety Analysis Report Markup (For Information Only), Revision 1

These revised Attachments 1, 2, and 3 supersede those submitted in Reference 1 Attachments 3, 4, and 5, respectively. The revised Updated Final Safety Analysis Report Sections contained in Attachment 4 supersede those previously contained in Reference 1 Attachment 8.

This information does not affect the results of the technical evaluation or the no significant hazards consideration determination previously transmitted in References 1 and 2.

PG&E makes no new or revised regulatory commitments (as defined by NEI 99-04) in this letter.

If you have any questions, or require additional information, please contact Mr. Hossein Hamzehee at (805) 545-4720.

I state under penalty of perjury that the foregoing is true and correct.

Executed on December 17, 2015.

Sincerely,

Edward D. Halpin
Senior Vice President – Power Generation and Chief Nuclear Officer

kjse/4328/50705089
Enclosure
cc: Diablo Distribution
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John P. Reynoso, Acting NRC Senior Resident Inspector
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Gonzalo L. Perez, Branch Chief, California Dept of Public Health

PG&E Response to NRC Request for Additional Information (RAI) Regarding License Amendment Request 15-03, "Application of Alternative Source Term"

NRC SRXB-RAI-1

Table 1 in supplement letter dated August 31, 2015, contains the current licensing basis (CLB) summary for the AST thermal hydraulic safety analyses for DCPP. The UFSAR [Updated Final Safety Analysis Report] accidents listed are loss-of-coolant accident (LOCA) peak cladding temperature [PCT], LOCA containment response, locked rotor accident [LRA], control rod ejection accident [CREA], main steam line break (MSLB) steam releases, steam generator tube rupture (SGTR), and loss of load (LOL) steam releases. For each UFSAR accident, the following items are listed per Table 1: source document, implemented by license amendment or 10 CFR 50.59, analysis date, and code methodology. The CLB for MSLB steam releases, SGTR, and LOL steam releases were done under the 10 CFR 50.59 process. The source documentation references two topical reports, WCAP-16638-P, "Diablo Canyon Units 1 and 2 Replacement Steam Generator Program NSSS [Nuclear Steam Supply System] Licensing Report," Revision 1, dated January 2008, and WCAP-16985-P, "Diablo Canyon Units 1 and 2 T_{avg} and T_{feed} Ranges Program NSSS Engineering Report," Revision 2, dated April 2009. The NRC staff did not review these topical reports with the implementation being done under the 10 CFR 50.59 process. The NRC staff requests that the licensee provide WCAP-16638-P, Revision 1, and WCAP-16985-P, Revision 2, for information for review of the applicable UFSAR Chapter 15 analyses thermal hydraulic parameter changes in support of the adoption of the proposed AST. Additionally, the NRC staff requests that the licensee provide calculation documents in support of these UFSAR Chapter 15 analyses in support of the adoption of the proposed AST. This information can be made available for NRC staff inspection via an audit.

PG&E Response

Pacific Gas and Electric Company (PG&E) has obtained agreement from Westinghouse, LLC, the owner of topical reports WCAP-16638-P and WCAP-16985-P and the referenced UFSAR Chapter 15 analysis calculations, to make the topical reports and UFSAR Chapter 15 analysis calculations available to the NRC for audit at the Westinghouse Rockville, MD, office on January 12 through January 14, 2016.

NRC SRXB-RAI-2

Section 4.3, "Gap Fractions for Non-LOCA Events," of Attachment 4 of the License Amendment Request [LAR] contains the information for the application regarding the gas gap fractions for the proposed AST for DCPP. The licensee states that the referenced gap fractions are contingent upon meeting Note 11 of Regulatory Guide [RG] 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Note 11 states the following:

> *The release fractions listed here have been determined to be acceptable for use with currently approved LWR [light-water reactor] fuel with a peak burnup of 62,000 MWD [megawatt days]/MTU [metric ton of uranium] provided that the maximum linear heat generation rate does not exceed 6.3 kw [kilowatts]/ft [foot] peak rod average power for burnups exceeding 54 GWD [gigawatt days]/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR [boiling-water reactor] rod drop accident and the PWR [pressurized-water reactor] rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.*

In its review of the gas gap fractions for the UFSAR Chapter 15 analyses, the NRC staff notes that the licensee is not calculating fission gas releases based on a bounding power history. Instead, the licensee is proposing new gas gap fractions selected from NUREG/CR 5009 and Safety Guide 25 without any analyses confirming that the values are bounding. The NRC staff requests that the licensee demonstrate that the new gas gap fractions are bounding based on fission gas release calculations performed using NRC-approved methodologies and using a bounding power history. Please include all information used to determine the new gas gap fractions for the DCPP AST. This information may include, but is not limited to, the following: analyses performed, methods used, operational power histories, input parameters, and assumptions or calculation bases applied.

PG&E Response

Subsequent to receipt of SRXB RAI-2 and during the related PG&E/NRC conference call dated October 13, 2015, it was agreed that in lieu of responding to the NRC request for "fission gas release calculations using NRC approved methods and bounding power history" to demonstrate that use of the highest fuel gap fractions per isotope/isotope class provided in NUREG/CR 5009, Safety Guide 25, and RG 1.183, Revision 0 (R0) was conservative. Furthermore, PG&E could instead use the fuel activity gap fractions from Table 3 of Draft Guide (DG)-1199 for non-LOCA events that experience fuel damage. This latter option was acceptable to NRC if PG&E could demonstrate that Diablo Canyon Power Plant (DCPP) falls within, and intends to operate within, the maximum allowable power operating envelope for pressurized water reactors (PWRs) shown in Figure 1 of DG-1199.

DCPP has three design basis non-LOCA accidents that are postulated to result in fuel damage, i.e., locked rotor accident (LRA), fuel handling accident (FHA), and control rod ejection accident (CREA). This change in gap fraction would affect the dose consequences reported in LAR 15-03 for the LRA and the FHA. It was agreed during the conference call between PG&E and the NRC, dated October 13, 2015, that per RG 1.183, R0, Note 11 of Table 3 and Appendix H, the dose consequences associated with the CREA remain unaffected by this request for additional information (RAI).

Compliance with Figure 1 of DG-1199

During normal operation, PG&E performs routine power distribution monitoring of the DCPP reactor core. Part of the power distribution surveillance requires the determination of the heat flux hot channel factor, F_Q. F_Q is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux based on nominal fuel pellet and fuel rod parameters. F_Q is directly related to the peak linear heat generation rate (LHGR). Peak LHGR can be determined by multiplying the average LHGR by F_Q.

Evaluation of the flux map data from Cycles 17, 18, and 19 of both units indicates that DCPP operates well-within the limits identified in Figure 1 of DG-1199. It is concluded that the current power peaking surveillance limit of 2.58 coupled with current DCPP core design practices provide a suitable limit for meeting the maximum allowable power operating envelope for PWRs shown in Figure 1 of DG-1199. For DCPP, the peak surveillance F_Q limit of 2.58 corresponds to a peak LHGR of 14.05 kW/ft.

To ensure future compliance with Figure 1 of DG-1199, and as part of the implementation of the amendment,

- DCPP Administrative Procedure TS6.DC3, "Reload Core Design Process," will be updated to include additional verification of core power peaking. Specifically, the limit on the peak rod LHGR in fuel assemblies during normal operation will be confirmed to remain within the nodal power envelope depicted for PWRs in Figure 1 of DG-1199.

- Routine surveillances performed during normal operation will verify that DCPP operates with sufficient margin to the current power peaking surveillance limit of 2.58 such that the peak rod LHGR history in the fuel assemblies in the DCPP core will remain within the nodal power envelope depicted for PWRs in Figure 1 of DG-1199.

Revised Fuel Gap fractions for Non-LOCA Events (specifically, LRA and FHA)

The activity gap fractions used in the DCPP dose consequence assessments for non-LOCA events FHA and LRA are intended to support flexibility in future DCPP fuel management schemes and address fuel rods that may exceed the RG 1.183, R0 LHGR criteria. PG&E is revising the fuel gap activity fractions used for non-LOCA events (specifically the LRA and FHA) in LAR 15-03 from the "highest fuel gap fractions per isotope/isotope class provided in NUREG/CR 5009, Safety Guide 25, and RG 1.183 R0," to those provided in "Table 3 of DG-1199." The above change reflects the conclusions of the conference call between PG&E and the NRC dated, October 13, 2015. DCPP falls within, and intends to operate within, the maximum allowable power operating envelope for PWRs shown in Figure 1 of DG-1199.

Table 1 below presents the difference between the gap fractions used in the FHA and LRA dose consequence analyses supporting the original submittal of LAR 15-03 (dated

June 17, 2015), versus the revised dose consequence analyses supporting the RAI response herein.

Table 1			
Group	**Gap Fraction**		**Comments**
	Original Analyses	Revised Analyses	
	NUREG/CR 5009, Safety Guide 25, and RG 1.183, R0	DG-1199	
I-131	0.12	0.08	Controlling dose contributor
I-132	0.10	0.23	Not a significant contributor to the total effective dose equivalent (TEDE) dose
Kr-85	0.30	0.35	Essentially a beta emitter, insignificant contributor to the TEDE dose
Other Noble Gases	0.10	0.04	Important for initiation of the Control Room (CR) radiation monitor signal to activate CR Mode 4 operation, minor contributor to the TEDE dose
Other Halogens	0.10	0.05	
Alkali Metals	0.17	0.46	• Does not affect dose consequences from the FHA since the alkali metals are retained in the pool • Does not significantly impact the LRA dose due to the very small steam generator moisture carry-over fraction (0.05%) for particulates

Revised Dose Consequences for the FHA and LRA

Table 2 below presents a comparison of the dose consequences for the following non-LOCA events using the "highest fuel gap fractions per isotope/isotope class provided in NUREG/CR 5009, Safety Guide 25, and RG 1.183, R0" (called "original" in the table below) and the revised gap fractions based on "Table 3 of DG-1199" (called "revised" in the table below).

- FHA in the Fuel Handling Building (FHB)
- FHA in the Containment
- LRA

It is concluded that at DCPP, use of fuel activity gap fractions from DG-1199 Table 3 (versus use of the highest fuel gap fractions per isotope/isotope class provided in NUREG/CR 5009, Safety Guide 25, and RG 1.183, R0) results in a slight reduction in the dose consequences following a FHA or a LRA.

Table 2							
	Reg. Limit Rem (TEDE)	FHA in the FHB Rem (TEDE)		FHA in Containment Rem (TEDE)		LRA Rem (TEDE)	
		Original	Revised	Original	Revised	Original	Revised
2-hr EAB	FHA= 6.3 LRA= 2.5	1.5	**1.0**	1.5	**1.0**	0.8	**0.5**
30-day LPZ	FHA= 6.3 LRA= 2.5	0.2	**0.1**	0.2	**0.1**	0.2	**0.1**
30-day CR	5	1.1	**1.0**	4.7	**4.3**	2.4	**1.7**

Revisions to the Attachments previously submitted as Attachments 3, 4, 5, and 8 of the Enclosure of LAR 15-03 to reflect the change in non-LOCA fuel activity gap fractions from "the highest fuel gap fractions per isotope/isotope class provided in NUREG/CR 5009, Safety Guide 25, and RG 1.183, R0", to those provided in "Table 3 of DG-1199," are contained in the following Attachments in the Enclosure to this letter:

- Attachment 1 – Technical Specification Bases Markup (For Information Only), Revision 1,
- Attachment 2 – Diablo Canyon Power Plant Technical Report Prepared by CB&I Stone and Webster, Inc., "Implementation of Alternative Source Terms, Summary of Dose Analyses and Results," Revision 1,
- Attachment 3 – Regulatory Guide 1.183 Conformance Tables, Revision 1, and
- Attachment 4 – Diablo Canyon Power Plant Updated Final Safety Analysis Report Markup (For Information Only), Revision 1 - The revised UFSAR Sections contained in Attachment 4 supersede those previously contained in Reference 1 Attachment 8. (Section 2.3, Section 15.5, and Tables from Section 15.5 supersede those previously contained in Reference 1 Attachment 8. Figure 2.3-5 is resubmitted to include the Title Block. Additionally, Tables from Section 6.0, which were inadvertently omitted in Reference 1 Attachment 8, are included in this submittal.)

License Amendment Request 15-03, Attachment 1

Technical Specification Bases Markup (For Information Only), Revision 1

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components were hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR ~~100~~50.67, "~~Reactor Site Criteria~~Accident Source Term" (Ref. 4).

(continued)

BASES (continued)

APPLICABILITY

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

SAFETY LIMIT VIOLATIONS

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

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR ~~100~~50.67, "~~Reactor Site Criteria~~Accident Source Term" limits (Ref. 4).

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

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14 (associated with 1967 GDC 9 per FSAR Appendix 3.1A), GDC 15 (no direct correlation to 1967 GDC; however, intent of 1971 GDC is per met per FSAR Appendix 3.1A), and GDC 28 (associated with 1967 GDC 30 per FSAR Appendix 3.1A).
2. ASME, Boiler and Pressure Vessel Code, Section III, Summer 1969.
3. ASME, Boiler and Pressure Vessel Code, Section XI.
4. 10 CFR ~~100~~50.67.
5. FSAR, Section 7.2.
6. DCM S-7, 3.4.1.

APPLICABLE SAFETY ANALYSIS (continued)

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

The startup of an inactive RCP in MODES 1 or 2 is precluded. In MODE 3, the startup of an inactive RCP cannot result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent start is less than half the minimum required SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

SDM satisfies Criterion 2 of 10CFR50.36(c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR ~~100~~50.67, "~~Reactor Site Criteria~~Accident Source Term," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be sufficient. The required SDM is specified in the COLR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.1.1.1 (continued) Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.
REFERENCES	1 10 CFR 50, Appendix A, GDC 26. 2 FSAR, Chapter 15, Section 15.4.2.1. 3 FSAR, Chapter 15, Section 15.2.4. 4 10 CFR ~~100~~50.67. 5 FSAR, Chapter 15, Section 15.4.6.1.6.

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

The LSSS, as defined in 10 CFR 50.36, are defined in this specification as the Allowable Values, and in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During AOOs, which are those events expected to occur more than once during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);
2. Fuel centerline melt shall not occur; and
3. The RCS pressure SL of 2735 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR ~~100~~50.67 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR ~~100~~50.67 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

(continued)

BASES

BACKGROUND (continued)

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB).
2. Fuel centerline melt shall not occur, and
3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR ~~100~~50.67 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR ~~100~~50.67 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable consequences for that event is considered having acceptable consequences for that event. However, these values and their associated NTSPs are not considered to be LSSS as defined in 10 CFR 50.36.

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below:

- Field transmitters or process sensors and instrumentation: provide a measurable electronic signal based on the physical characteristics of the parameter being measured;
- Signal processing equipment including digital protection system, field contacts, and protection channel sets: provide signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications; and
- Solid State Protection System (SSPS) including input, logic, and output bays: initiates the proper unit shutdown or engineered safety feature (ESF) actuation in accordance with the defined logic and based on the bistable outputs from the signal process control and protection system. The residual heat removal pump trip or refueling water storage tank level-low signal is not processed by the SSPS. The associated relays are located in the residual heat removal pumps control system.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

They are also the primary means for automatically isolating containment in the event of a fuel handling accident or any other source within containment during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR ~~100~~50.67 (Ref. 1) limits. Due to radioactive decay, containment is only required to isolate during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours.)

The containment ventilation isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Ventilation Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation - Not used

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI, and ESFAS Function 3.a, Containment Phase A Isolation. The applicable MODES and specified conditions for the Containment Ventilation Isolation portion of these Functions are different and less restrictive than those for their Phase A isolation and SI roles. If one or more of the SI or Phase A isolation Functions becomes inoperable in such a manner that only the Containment Ventilation Isolation Function is affected, the Conditions applicable to their SI and Phase A isolation Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Ventilation Isolation Functions specify sufficient compensatory measures for this case.

3. Containment Radiation

The LCO specifies two required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment ventilation Isolation remains OPERABLE in MODES 1-4.

The LCO only requires one monitor to be OPERABLE during movement of recently irradiated fuel assemblies in containment. In order to provide the CVI function under these conditions without placing the entire SSPS in service, an alternate circuit is provided to power the output relays and provide logic actuation signals independent of the SSPS.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.7 (continued)

The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.6.8

This SR assures that the individual channel RESPONSE TIMES for the CVI from Containment Purge Radiation Gaseous and Particulate function are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in ECG 38.2. Individual component response times are not modeled in the analyses. The analyses model the overall or elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., valves in full closed position). The response time may be measured by a series of overlapping tests such that the entire response time is measured.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR ~~100.11~~50.67.
2. NUREG-1366, December 1992.
3. DCM No. T-16, Containment Function.
4. WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," March 2003.
5. License Amendment 184/186, January 3, 2006.

BASES

BACKGROUND (continued)

The CRVS has two additional manually selected emergency operating modes; smoke removal and recirculation. Neither of these modes are required for the CRVS to be OPERABLE, but they are useful for certain non-DBA circumstances.

APPLICABLE SAFETY ANALYSES

The control room must be kept habitable for the operators stationed there during accident recovery and post accident operations.

The CRVS acts to terminate the supply of unfiltered outside air to the control room, initiate filtration, and pressurize the control room. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post accident operations by minimizing the radiation exposure of control room personnel.

In MODES 1, 2, 3, and 4, the radiation monitor (located at the control room intakes) actuation of the CRVS is a backup for the Phase A signal actuation. This ensures initiation of the CRVS during a loss of coolant accident, ~~or~~ steam generator tube rupture, control rod ejection accident and Main Steam Line Break ~~involving a release of radioactive materials~~.

The radiation monitor actuation of the CRVS in MODES 5 and 6, during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours), is the primary means to ensure control room habitability in the event of a fuel handling or waste gas decay tank rupture accident. This actuation is credited in the FHA. The CRVS pressurization system actuation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

In MODES 1, 2, 3, 4, 5 and 6, credit is taken for the dual ventilation intake design of the CR pressurization air intakes. Based on the availability of redundant PG&E Design Class I radiation monitors at each pressurization intake location, the DCPP design has the capability of initial selection of the cleaner intake, but does not have the capability of automatic selection of the clean intake throughout the event. Based on the CRVS pressurization intake design, and the expectation that the operator will manually make the proper intake selection throughout the event, and per RG 1.194, June 2003, Regulatory Position C.3.3.2.3, when the CRVS is in Mode 4, the X/Q values for the more favorable CR intake is reduced by a factor of 4 and utilized to estimate the dose consequences.

LCO

The LCO requirements ensure that instrumentation necessary to initiate the CRVS pressurization system is OPERABLE.

1. Manual Initiation

   The LCO requires two trains OPERABLE. The operator can initiate the CRVS pressurization mode at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

2. Automatic Actuation Relays

The LCO requires two trains of Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation of the pressurization system. Since each unit has one train of Actuation Relays consisting of two sets of actuation logic, each unit must have at least one logic set for both trains to be considered OPERABLE.

(continued)

(Spillover from previous page.)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.5

SR 3.3.7.5 is the performance of a SLAVE RELAY TEST. This test energizes the Slave Relays and verifies actuation of the equipment to the pressurization mode. Although there are no "Slave Relays" as in the SSPS, this surveillance was maintained to preserve the format of the standard specification. The surveillance is intended to ensure that the actuation relays, downstream of the logic, function to actuate the pressurization mode equipment. Since the radiation monitors directly actuate the actuation relays, this test is performed as part of the performance of SR 3.3.7.2.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.7.6

SR 3.3.7.6 is the performance of a TADOT. This test is a check of the Manual Actuation Functions. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

SR 3.3.7.7

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

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. WCAP-13878, "Reliability of Potter & Brumfield MDR Relays", June 1994.
2. WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals", April 1994.
3. License Amendment 184/186, January 3, 2006.
4. RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003.

B 3.3 INSTRUMENTATION

B 3.3.8 Fuel Building Ventilation System (FBVS) Actuation Instrumentation

BASES

BACKGROUND

The FBVS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours) are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.13, "Fuel Handling Building Ventilation System." The system initiates filtered ventilation of the fuel building automatically following receipt of a high radiation signal from the Spent Fuel Pool Monitor or from the New Fuel Storage Vault Monitor. Initiation may also be performed manually as needed from the main control room or fuel handling building.

High radiation, from either of the two monitors, provides FBVS initiation. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the fuel building.

APPLICABLE SAFETY ANALYSES

The FBVS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident involving handling recently irradiated fuel are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the fuel building exhaust following a fuel handling accident ~~so that offsite doses remain within the limits specified in 10 CFR 100 (Ref. 1)~~.

The FBVS actuation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that instrumentation necessary to initiate the FBVS is OPERABLE.

1. <u>Manual Initiation</u>

   The LCO requires two channels OPERABLE. The operator can initiate the FBVS at any time by using either of two switches, one in the control room and another in the fuel handling building. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

   The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1 (continued)

The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.8.2

A CFT is performed on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the FBACS actuation. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.3.8.3 - Not used

SR 3.3.8.4

SR 3.3.8.4 is the performance of a TADOT. This test is a check of the manual actuation functions. Each manual actuation function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (e.g., pump starts, valve cycles, etc.). The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

SR 3.3.8.5

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. ~~10 CFR 100.11~~Not used.
2. License Amendment 184/186, January 3, 2006.
3. PG&E Letter DCL-05-124

BASES (continued)

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. Safety analyses for design basis events that model primary to secondary LEAKAGE result in steam discharge to the atmosphere. ~~The safety analysis for the SLB event assumes that primary to secondary LEAKAGE is 10.5 gpm (room temperature conditions) from the faulted SG or increases to 10.5 gpm as a result of accident induced conditions, and 0.1 gpm (room temperature conditions) from each intact SG.~~ The safety analyses for events resulting in steam discharge to the atmosphere~~,~~ ~~other than SGTR and SLB,~~ assume that primary to secondary LEAKAGE from all SGs is 0.75 gpm (~~hot conditions~~Standard Temperature and Pressure) under accident conditions. For conservatism, the SLB assumes that the total 0.75 gpm tube leakage is assigned to the faulted steam generator and the SGTR assumes that the total 0.75 gpm tube leakage is assigned to the 3 intact steam generators. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the SLB safety analysis for the faulted SG.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The SGTR (Ref. 3) is more limiting for radiological releases at the site boundary. The radiological dose analysis assumes loss of off-site power at the time of reactor trip with no subsequent condenser cooling available. The SGTR assumes that the total 0.75 gpm tube leakage is assigned to the 3 intact steam generators. The steam generator (SG) PORV for the SG that has sustained the tube rupture is assumed to fail open for 30 minutes, at which time the operator closes the block valve to the PORV. The dose consequences resulting from the SGTR accident are within the limits defined in 10 CFR ~~100~~50.67 (Ref. 6).

The SLB is more limiting for site radiation releases for events other than SGTR. ~~The safety analysis for the SLB accident assumes 10.5 gpm primary to secondary LEAKAGE is through the faulted SG.~~ The dose consequences resulting from the SLB accident are ~~well~~ within the limits defined in 10 CFR ~~100~~50.67~~ or the staff approved licensing basis (i.e., small fraction of these limits)~~.

The safety analysis for RCS main loop piping for GDC-4 (Ref. 1) assumes 1 gpm unidentified leakage and monitoring per RG 1.45 (Ref. 2) are maintained (Ref. 4 and 5).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 4 and 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 15.
4. FSAR, Section 3.
5. NUREG-1061, Volume 3, November, 1984.
6. 10 CFR ~~100~~50.67.
7. NEI 97-06, “Steam Generator Program Guidelines.”
8. EPRI, “Pressurized Water Reactor Primary-to-Secondary Leak Guidelines.”

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose ~~to the whole body and the thyroid~~ that an individual at the exclusion area boundary can receive for 2 hours following an accident or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR ~~100.11~~50.67 (Ref. 1). Doses to the control room operators must be limited per GDC 19. The limits on specific activity ensure that the doses are appropriately limited during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in ~~the Standard Review Plan~~ RG 1.183 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following a SLB or a SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at ~~or more conservative than~~ the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of ~~1~~0.75 gpm~~ exists~~. The safety analyses assume the specific activity of the secondary coolant is at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.18, "Secondary Specific Activity."

The analysis for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analyses consider two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a SLB (by a factor of 500) or SGTR (by a factor of 335), respectively.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas specific activity is assumed to be ~~651~~270 μCi/gm DOSE EQUIVALENT XE-133.

The SGTR analysis also assumes a loss of offsite power at the same time as the reactor trip. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR system is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed 60.0 μCi/gm DOSE EQUIVALENT I-131, for more than 48 hours.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The iodine specific activity in the reactor coolant is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to ~~600.0~~270.0 μCi/gm DOSE EQUIVALENT XE-133, as contained in SR 3.4.16.2 and SR 3.4.16.1 respectively. The limits on specific activity ensure that offsite and control room doses will meet the appropriate ~~SRP~~ acceptance criteria (Refs. 1 and 2).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

The definition of DOSE EQUIVALENT XE-133 in Specification 1.1, "Definitions," requires that the determination of DOSE EQUIVALENT XE-133 shall be performed using the effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil." These dose conversion factors are consistent with the dose conversion factors used in the applicable dose consequence analyses.

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

SR 3.4.16.2

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

The definition of DOSE EQUIVALENT I-131 in Specification 1.1, "Definitions," specifies the ~~thyroid~~ dose conversion factors which may be used to determine DOSE EQUIVALENT I-131. The ~~thyroid~~ dose conversion factors used to determine DOSE EQUIVALENT I-131 are the committed thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," and are~~to be~~ consistent with the dose conversion factors used in the applicable dose consequence analyses~~, or be conservative with respect to the dose conversion factors used in the applicable dose consequence analyses such that a higher DOSE EQUIVALENT I-131 is determined~~.

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR ~~100.11, 1973~~50.67.
2. ~~Standard Review Plan (SRP), Section 6.4 (SLB and SGTR control room dose limits), Section 15.1.5 Appendix A (SLB offsite dose limits) and Section 15.6.3 (SGTR offsite dose limits).~~Regulatory Guide 1.183, July 2000.
3. FSAR, Sections 15.4.3 and 15.5.20.
4. FSAR Section ~~15.1.5~~15.5.18.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a ~~total~~ primary to secondary LEAKAGE rate of ~~1~~0.75 gpm from the intact SGs plus the leakage rate associated with a double-ended rupture of a single tube. The SGTR radiological dose analysis assumes loss of off-site power at the time of reactor trip with no subsequent condenser cooling available. The SG PORV for the SG that has sustained the tube rupture is assumed to fail open for 30 minutes, at which time the operator closes the block valve to the PORV. The SGTR radiological dose analysis assumes the contaminated secondary fluid is released briefly to the atmosphere from all the PORVs following reactor trip, is released from the ruptured SG PORV for 30 minutes, is released from the intact SG PORVs during the cooldown, and is released from all PORVs following cooldown until termination of the event.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) ~~For the SLB event, the primary to secondary LEAKAGE is 10.5 gpm from the faulted SG or is assumed to increase to 10.5 gpm as a result of accident induced conditions, and 0.1 gpm from each intact SG. For other events, the~~The steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 0.75 gpm under accident 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 3.4.16, "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 ~~100~~50.67 (Ref. 3) ~~or the NRC approved licensing basis (e.g., a small fraction of these limits)~~ and RG 1.183 (Ref. 7).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG 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.

(continued)

BASES

LCO (continued)

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 (a) that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions, and (b) that the primary to secondary LEAKAGE will not exceed ~~1~~0.75 gpm total for all four ~~per~~ SGs (except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage) to ensure that the potential for induced leakage during severe accidents will be maintained at a level that will not increase risk. The accident analysis for the SLB event, the SGTR event and other events resulting in steam release to the atmosphere assumes that accident induced leakage does not ~~exceed 10 gpm in the faulted SG and 0.1 gpm in each intact SG. For the faulted SG in the SLB event, 10.5 gpm is the accident induced leakage limit, of which no more than 1 gpm can come from sources not specifically exempted by the NRC from this 1 gpm limit. The accident analyses for events other than SGTR and SLB assume that leakage does not~~ exceed 0.75 gpm total under accident conditions. 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 SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG 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.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

In addition, Specification 5.5.9 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.17.2

During an SG 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 5.5.9 are intended to ensure that tubes accepted for continued service satisfy the SG 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 SG 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 SG performance criteria.

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG 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, 1999.
3. 10 CFR ~~100~~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."
7. Regulatory Guide 1.183, July 2000.

BASES

BACKGROUND (continued)

Containment Purge System (48 inch purge valves)

The Containment Purge System operates to supply outside air into the containment for ventilation and cooling or heating needed for prolonged containment access following a shutdown and during refueling. ~~The system may also be used to reduce the concentration of noble gases within containment prior to and during personnel access.~~ The supply and exhaust lines each contain two isolation valves. ~~The 48 inch Containment Purge valves are qualified for automatic closure from their open position under DBA conditions.~~ The safety analyses assume that the 48-inch supply and exhaust line valves are closed at the start of the DBA. Therefore, the 48 inch Containment Purge supply and exhaust isolation valves are ~~normally maintained~~ sealed closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained. ~~The Containment Purge Supply and Exhaust Isolation valves are supplied with an internal block which prevents opening the valve beyond 80 degrees. This block was provided by the manufacture to allow limiting the valve's opening. Calculations performed during qualification to Branch Technical Position CSB 6-4 showed the block to be unnecessary to assure closure time within 2 seconds under DBA conditions (SSER 9, June 1980 and Calculation M-661). Adjustments of this block to values greater than or less than 80 degrees will not affect the valve's ability to close. This design assures that containment boundary is maintained. These valves may be opened as necessary to:~~

~~a. Reduce noble gases within containment prior to and during personnel access, and~~

~~b. Mitigate the effects of controller leakage and other sources which may effect the habitability of the containment for personnel entry.~~

~~Operation in Modes 1, 2, 3, or 4 with the 48-inch purge valves or the 12-inch vacuum/pressure relief valves open providing a flow path is limited to no more than 200 hours per calendar year.~~

Containment Pressure/Vacuum Relief (12 inch isolation valves)

The Containment Pressure/Vacuum Relief valves are operated as necessary to:

a. Reduce the concentration of noble gases within containment prior to and during personnel access, and

b. Equalize containment internal and external pressures.

Since the 12 inch Containment Pressure/Vacuum Relief valves are designed to meet the requirements for automatic containment isolation within 5 seconds if mechanical blocks are installed to prevent opening more then 50°, these valves may be opened as needed in MODES 1, 2, 3, and 4.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBA that results in a release of radioactive material within containment in MODES 1, 2, 3, or 4 is a loss of coolant accident (LOCA) (Ref. 1). In the analyses for this accident, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including the ~~Containment Purge, and~~ Containment Vacuum/Pressure Relief valves) are minimized. The safety analyses assume that the 48 inch purge valves are closed at event initiation. If the ~~48 inch Containment Purge supply and exhaust valves close within 2 seconds and the~~ 12 inch pressure/vacuum relief valves close within 5 seconds after the DBA initiation, the safety analysis shows that offsite dose release will be less than 10 CFR~~100~~ 50.67 guidelines.

The DBA analysis assumes that containment isolation occurs and leakage is prevented except for the design leakage rate, L_a.

The LOCA offsite dose analysis assumes leakage from the containment at a maximum leak rate of 0.10 percent of the containment volume per day for the first 24 hours, and at 0.05 percent of the containment volume per day for the duration of the accident.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of ~~the 48 inch Containment Purge supply and exhaust and~~ the 12 inch Containment Pressure/Vacuum Relief valves. Two valves in series provide assurance that the flow paths can be isolated even if a single failure occurred. The inboard and outboard isolation valves are provided with diverse power sources and are pneumatically operated spring closed valves that will fail closed on the loss of power or air.

The ~~48 inch Containment Purge supply and exhaust and~~ 12 inch Containment Pressure/Vacuum Relief valves are able to close in the environment following a LOCA. Therefore, each of the ~~Containment Purge supply and exhaust and~~ Containment Vacuum/pressure Relief valves may be opened to provide a flow path. The ~~48 inch Containment Purge supply and exhaust valves and/or~~ 12-inch vacuum/pressure relief valves may be open no more then 200 hours per calendar year while in MODES 1, 2, 3, and 4. ~~Additionally, only two of the three flow paths (containment purge supply and exhaust, and containment vacuum/pressure relief) may be open at one time.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The system is designed to preclude a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.

The 48 inch Containment Purge supply and exhaust valves may be unable to close in the environment following a LOCA in sufficient time to support DBA acceptance criteria. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. In this case, the single failure criterion remains applicable to the containment purge valves due to failure in the control circuit associated with each valve. Again, the purge system valve design precludes a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.

The containment isolation valves satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA. The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 48 inch Containment Purge supply and exhaust valves ~~and the~~ must be sealed closed during MODES 1, 2, 3, and 4. The Pressure/Vacuum Relief valves must have blocks installed to prevent full opening. These blocked valves also actuate on an automatic isolation signal. The valves covered by this LCO are listed along with their associated stroke times in Plant Procedure AD13.DC1 (Ref. 5).

Normally closed passive containment isolation valves/devices are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 5.

Containment Purge supply and exhaust valves, and Containment Pressure/Vacuum Relief valves with resilient seals must meet additional leakage rate surveillance frequency requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment."

This LCO provides assurance that the containment isolation valves and the Containment Purge supply and exhaust, and Containment Pressure/Vacuum Relief valves will perform their designed safety function to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

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BASES (continued)

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, except 48-inch purge valve flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a person at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow path containing these valves may not be opened under administrative controls. A single purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1.~~This Note also limits operation of the normally isolated Containment Supply and Exhaust valves (2 penetration flow paths) and the Vacuum/Pressure Relief valves (1 penetration flow path) to no more than 2 of 3 penetration flow paths open at one time.~~

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

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

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

Plant Procedure AD13.DC1 Attachment 7.7 (Ref. 5) provides the applicable CONDITION to enter for each containment isolation valve if the valve is inoperable.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths requiring isolation following a DBA is inoperable except for Containment Purge supply and exhaust, and Containment Pressure/Vacuum Relief isolation valve leakage not within limit, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that

(continued)

BASES

ACTIONS

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

condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per 92 days was chosen and has been shown to be acceptable based on operating experience.

Required Action D.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

E.1 and E.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

~~Not Used~~ Each 48 inch Containment Purge supply and exhaust valve is required to be verified sealed closed. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a Containment Purge valve. These valves are assumed to be closed at the start of a DBA. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A Containment Purge valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator. In the event the purge valve leakage requires entry into Condition D, the surveillance permits opening one purge valve in a penetration flow path to perform repairs. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.6.3.2

This SR ensures that ~~the 48 inch Containment Purge supply and exhaust and~~ the 12 inch Containment Pressure/Vacuum Relief valves are closed as required or, if open, open for an allowable reason. If a purge or pressure relief valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the ~~Containment Purge supply and exhaust or~~ Containment Pressure Relief valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The ~~Containment Purge supply and exhaust or~~ Containment Pressure/Vacuum Relief valves are capable of closing in the

(continued)

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BASES

BACKGROUND

Containment Spray System (continued)

In the recirculation mode of operation, containment spray is supplied by manual realignment of the residual heat removal (RHR) pumps after the RWST is empty.

The Containment Spray System provides a spray of cold borated water mixed with sodium hydroxide (NaOH) from the spray additive tank into the upper regions of containment to reduce the containment pressure and temperature, and to reduce fission products from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the RHR heat exchangers. Each train of the Containment Spray System provides adequate spray coverage to meet the system design requirements for containment atmospheric heat removal.

The Spray Additive System injects an NaOH solution into the spray. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water maximizes the retention of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The Containment Spray System is actuated either automatically by a containment High-High pressure signal or manually. If an "S" signal is present, the High-High pressure signal automatically starts the two containment spray pumps, opens the containment spray pump discharge valves, opens the spray additive tank outlet valves, initiates a phase "B" isolation signal, and begins the injection phase. A manual actuation of the Containment Spray System will begin the same sequence and can be initiated by operator action from the main control board. The injection phase of containment spray continues until an RWST Low-Low level alarm is received. The Low-Low level alarm for the RWST signals the operator to manually secure the system. After re-alignment of the RHR system to the containment recirculation sump, the associated RHR spray header isolation valve ~~may be~~is opened to allow continued spray operation of one train of spray utilizing the RHR pump to supply flow. The LOCA dose analysis takes credit for this manual initiation of Containment Spray during recirculation to take place within 12 minutes following the termination of Containment Spray during the injection phase.

Containment Spray is ~~not~~ required to be actuated during the recirculation phase of a LOCA~~, but may be actuated at the discretion of the Technical Support Center~~. Containment Spray operation (injection plus recirculation) is credited until 6.25 hours following

<u>initiation of a LOCA.</u> During the recirculation phase of a LOCA, the Containment Spray System must be capable of

(continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)	Analyses and evaluation show that containment spray is not required during the recirculation phase of a LOCA for containment pressure and temperature control (Ref. 7). However, for dose consequences, containment spray is required during the recirculation phase of a LOCA for removing radioactive iodine and particulates from the containment atmosphere. If only one RHR pump is available during the recirculation phase of a LOCA, it may not be possible to obtain significant containment spray without closing valves 8809A or B. If recirculation spray is used with only one train of RHR in operation, ECCS flow to the reactor will be reduced, but analysis has shown that the flow to the reactor in this situation is still in excess of that needed to supply the required core cooling. The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a -1.80 psid containment pressure decrease and is based on a sudden cooling effect of 70°F in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4. The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-High pressure setpoint to achieving full flow through the containment spray nozzles. The Containment Spray System total response time includes diesel generator (DG) startup (for loss of offsite power),

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	sequenced loading of equipment, containment spray pump startup, and spray line filling (Ref. 4). The CFCUs performance for post accident conditions is given in Reference 4. The result of the analysis is that each train (two CFCUs) combined with one train of containment spray can provide 100% of the required peak cooling capacity during the post accident condition. The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-High pressure setpoint to achieving full Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time includes signal delay, DG startup (for loss of offsite power), and component cooling water pump startup times. The Containment Spray System and the Containment Cooling System satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).
LCO	During a DBA LOCA, a minimum of two CFCUs and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits (Refs. 4). Additionally, one containment spray train is also required to remove radioactive iodine and particulates from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and the CFCU system consisting of four CFCUs or three CFCUs each supplied by a different vital bus must be OPERABLE. Therefore, in the event of an accident, at least one train of containment spray and two CFCUs operate, assuming the worst case single active failure occurs. Each Containment Spray train typically includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal. Upon actuation of the RWST Low-Low alarm, the containment spray pumps are secured. Containment spray ~~could~~is then ~~be~~ supplied ~~as required~~ by an RHR pump taking suction from the containment sump for a total spray operation (injection and recirculation) of 6.25 hours. Each CFCU includes cooling coils, dampers, fans, instruments, and controls to ensure an OPERABLE flow path.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains and CFCUs. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray System and the Containment Cooling System are not required to be OPERABLE in MODES 5 and 6.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs. d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases. e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned. The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	This LCO requires that four MSIVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal. This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR ~~100~~50.67 (Ref. 4) limits or the NRC staff approved licensing basis.
APPLICABILITY	The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed and de-activated (vented or prevented from opening), when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function. In MODE 4, the steam generator energy is low, thus OPERABILITY in MODE 4 is not required. In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.
ACTIONS	A.1 With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the MSIV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs. The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power.

As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program.

This test may be conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. However, the test is normally conducted in MODE 5 as permitted by the cold shutdown frequency justification provided in the Inservice Testing Program (IST) and as permitted by Reference 6, Subsection ISTC-3521(c).

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 10.3.
2. FSAR, Section 6, Appendix 6.2 D.
3. FSAR, Section 15.4.2.
4. 10 CFR ~~100.11~~50.67.
5. ASME, Boiler and Pressure Vessel Code, Section XI.
6. ASME Code for Operation and Maintenance of Nuclear Power Plants, 2001 Edition including 2002 and 2003 Addenda.

BASES

BACKGROUND (continued)

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CRVS is designed in accordance with Seismic Category I requirements.

The CRVS is designed to maintain a habitable environment in the CRE for the duration of the most severe Design Basis Accident (DBA) without exceeding a 5 rem ~~whole body~~TEDE dose ~~or its equivalent to any part of the body~~.

APPLICABLE SAFETY ANALYSES

The CRVS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The CRVS provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE occupant dose analyses for the most limiting design basis accident, fission product release presented in the FSAR, Chapter 15 (Ref. 2).

There are no offsite or onsite hazardous chemicals that would pose a credible threat to control room habitability. Consequently, engineered controls for the control room are not required to ensure habitability against a hazardous chemical threat. The amount of CRE unfiltered inleakage is not incorporated into PG&E's hazardous chemical assessment.

The evaluation of a smoke challenge demonstrated that smoke will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels (Ref. 1). The assessment verified that a fire or smoke event anywhere within the plant would not simultaneously render the Hot Shutdown Panel (HSP) and the CRE uninhabitable, nor would it prevent access from the CRE to the HSP in the event remote shutdown is required. No CRVS automatic actuation is required for hazardous chemical releases or smoke and no Surveillance Requirements are required to verify operability in cases of hazardous chemicals or smoke.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The worst case single active failure of a component of the CRVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. The CRVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant CRVS trains are required to be OPERABLE to ensure that at least one is available if a single active failure disables the other train. The redundant train means a second train from the other unit (Ref. 5). Total system failure, such as from a loss of both ventilation trains or from an inoperable CRE boundary, could result in exceeding a dose of 5 rem ~~whole body~~TEDE ~~or its equivalent to any part of the body~~ to the CRE occupants in the event of a large radioactive release.
Each CRVS train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. A CRVS train is OPERABLE when the associated:

a. main supply fan (one), filter booster fan (one) and pressurization fan (one) are OPERABLE;

b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In order for the CRVS trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs. In the event of an inoperable CRE boundary in MODES 1, 2, 3, or 4, mitigating actions are required to ensure CRE occupants are protected from hazardous chemicals and smoke.

DCPP does not have CRVS automatic actuation for hazardous chemicals or smoke. Current practices at DCPP do not utilize chemicals in sufficient quantity to present a chemical hazard to the control room. Smoke is not considered in the DCPP safety analyses. Therefore, there are no specific limits at DCPP for hazardous chemicals or smoke.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6, and during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours) the CRVS must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA or the release from the rupture of an outside waste gas tank.

During movement of recently irradiated fuel assemblies, the CRVS must be OPERABLE to cope with the release from a fuel handling accident involving handling recently irradiated fuel.

CRVS OPERABILITY requires that for MODE 5 and 6 and during movement of recently irradiated fuel assemblies in either unit, when there is only one OPERABLE train of CRVS, the OPERABLE CRVS train must be capable of being powered from an OPERABLE diesel generator that is directly associated with the bus which is energizing the OPERABLE CRVS train. This is an exception to LCO 3.0.6.

ACTIONS

The ACTIONS are modified by a NOTE that states that ACTIONS apply simultaneously to both units. The CRVS is common to both units.

A.1

When one CRVS train is inoperable for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CRVS train is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CRVS train could result in loss of CRVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

B.1, B.2, and B.3

The CRE boundary is inoperable if unfiltered inleakage past the CRE boundary can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem ~~whole body or its equivalent to any part of the body~~TEDE).

In the event of an inoperable CRE boundary in MODES 1, 2, 3, or 4, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from potential smoke and chemical hazards.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.3

This SR verifies that the required CRVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CRVS filter tests are in accordance with ANSI N510-1980 (Ref. 3). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.4

This SR verifies that each CRVS train automatically starts and operates in the pressurization mode on an actual or simulated actuation signal generated from a Phase "A" Isolation. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.7.10.5

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program. Any changes to the most limiting configuration of the CRVS testing alignment for determining unfiltered air inleakage past the CRE boundary into the CRE must be made using a conservative decision making process (References 11-13).

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem ~~whole body or its equivalent to any part of the body~~TEDE and the CRE occupants are protected from hazardous chemicals and smoke. For DCPP, there is no CRVS automatic actuation for hazardous chemical releases or smoke and there are no CRVS Surveillance Requirements that verify operability in cases of hazardous chemicals or smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident.

(continued)

BASES

BACKGROUND (continued)	The ABVS is discussed in the FSAR, Sections 9.4 2, and 15.5 (Refs. 1, and 2, respectively) since it may be used for normal, as well as post accident, ventilation and atmospheric cleanup functions. The primary purpose of the single manually initiated heater is to maintain the relative humidity at an acceptable level, consistent with iodine removal efficiencies per ASTM D 3803-1989 (Ref. 3). There is no redundant heater since the failure of the charcoal adsorber and heater train would constitute a second failure in addition to the RHR pump seal failure assumed in conjunction with a LOCA (Ref.7). The heaters are not required for ABVS operability.
APPLICABLE SAFETY ANALYSES	The design basis of the ABVS is established by the large break LOCA. The system evaluation assumes a passive failure of the ECCS outside containment, such as an RHR pump seal failure, during the recirculation mode. In such a case, the system limits radioactive release to within the 10 CFR ~~100~~50.67 (Ref. 5) limits. The analysis of the effects and consequences of a large break LOCA is presented in Reference 2. The ABVS also functions, following a LOCA, in those cases where the ECCS goes into the recirculation mode of long term cooling, to clean up releases of smaller leaks, such as from valve stem packing. The ventilation flow is also required to maintain the temperatures of the operating ECCS motors within allowable limits. The ventilation function has been designed for single failure and the system will continue to function to provide its ECCS motor cooling function. Two types of system failures are considered in the accident analysis for radiation release: complete loss of function of one train, and excessive RHR pump seal LEAKAGE. Either type of failure may result in a lower efficiency of removal for any gaseous and particulate activity released to the ECCS pump rooms following a LOCA. The ABVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Two trains of the ABVS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power. Total system failure could result in the atmospheric release from the ECCS pump room exceeding 10 CFR ~~100~~50.67 limits in the event of a Design Basis Accident (DBA). ABVS is considered OPERABLE when the individual components necessary to maintain the ECCS pump room filtration and temperature are OPERABLE in both trains.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.7.12.6 This SR verifies the leak tightness of dampers that isolate flow to the normally operating filter train. This SR assures that the flow from the auxiliary building passes through the HEPA filter and charcoal adsorber unit when the ABVS Buildings and Safeguards or Safeguards Only modes have been actuated coincident with an SI. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.
REFERENCES	1. FSAR, Section 9.4.2. 2. FSAR, Section 15.5. 3. ASTM D 3803-1989 4. ANSI N510-1980 5. 10 CFR ~~100.11~~50.67. 6. NUREG-0800, Section 6.5.1, Rev. 2, July 1981. 7. DCM S-23B, "Main Auxiliary Building Heating and Ventilation System".

BASES

APPLICABLE SAFETY ANALYSES (continued)	FHBVS is only required to isolate during fuel handling accidents involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours). In accordance with assumptions made in the fuel handling accident analysis, loss of offsite power is not considered concurrent with a fuel handling accident. ~~However, loss of power is enveloped by the fuel handling accident analysis.~~ To maximize FHBVS capability to mitigate the consequences of a fuel handling accident, at least one of the FHBVS trains must be capable of being supplied from an operable emergency diesel generator at all times whenever movement of recently irradiated fuel is taking place in the spent fuel pool. These assumptions and the analysis follow the guidance provided in Regulatory Guide ~~1.25~~1.183 (Ref. 3). The FHBVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Two independent and redundant trains of the FHBVS are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train. In accordance with assumptions made in the fuel handling accident analysis, loss of offsite power is not considered concurrent with a fuel handling accident. ~~However, loss of power is enveloped by the fuel handling accident analysis.~~ This requires that when two trains of the FHBVS are OPERABLE, at least one train of the FHBVS must be capable of being powered from an OPERABLE diesel generator that is directly associated with the bus which energizes the FHBVS train. When only one train is OPERABLE, an OPERABLE diesel generator must be directly associated with the bus which energizes that one OPERABLE FHBVS train. Total system failure could result in the atmospheric release from the fuel handling building exceeding the 10 CFR ~~100~~50.67 (Ref. 4) limits in the event of a fuel handling accident. The FHBVS is considered OPERABLE when the individual components necessary to control releases from fuel handling building are OPERABLE in both trains. An FHBVS train is considered OPERABLE when its associated: a. Exhaust fan is OPERABLE; b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.4

This SR verifies the integrity of the fuel handling building enclosure. The ability of the fuel handling building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FHBVS. During the post accident mode of operation, the FHBVS is designed to maintain a slight negative pressure in the fuel handling building, to prevent unfiltered LEAKAGE. The FHBVS is designed to maintain the building pressure ≤ -0.125 inches water gauge with respect to atmospheric pressure. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.7.13.5

Operation of damper M-29 is necessary to ensure that the system functions properly. The operability of damper M-29 is verified if it can be closed. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 9.4.4.
2. FSAR, Section 15.5.
3. Regulatory Guide ~~1.25~~1.183, July 2000.
4. 10 CFR ~~100~~50.67.
5. ASTM D 3802-1989
6. ANSI N510-1980.
7. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.
8. DCM S-23D, "Fuel handling Building HVAC System."
9. Not used
10. License Amendment 184/186, January 3, 2006.
11. PG&E Letter DCL-05-124

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the spent fuel pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. A general description of the spent fuel pool design is given in the FSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section 9.1.4.3.4, 15.4.5 and 15.5.22 (Ref. 3).
APPLICABLE SAFETY ANALYSES	The minimum water level in the spent fuel pool meets the assumptions of the fuel handling accident described in Regulatory Guide ~~1.25~~1.183 (Ref. 4). The resultant 2 hour ~~thyroid~~ dose per person at the exclusion area boundary is a small fraction of the 10 CFR ~~100~~50.67 (Ref. 5) limits. According to Reference 4, there is 23 ft of water between the top of the damaged fuel rods and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. Although there are other spent fuel pool elevations where fuel handling accidents can occur, the design basis fuel handling accident, which uses the conservative assumptions of RG ~~1.25~~1.183, is expected to be bounding. To add conservatism, the analysis assumes that all fuel rods of the damaged fuel assembly fail. In practice, the water level maintained for fuel handling provides more than 23 feet of water over the top of irradiated fuel assemblies seated in the storage racks. FSAR Section 9.1.4.3.4 requires the water level provide a minimum of 8 feet of water shielding during fuel handling. This assures more than 24 feet 6 inches of water shielding over the top of the fuel assemblies in the racks and more than 23 feet of water shielding over a fuel assembly lying horizontally on top of the racks. The spent fuel pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The spent fuel pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

(continued)

BASES (continued)

APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the spent fuel pool, since the potential for a release of fission products exists.
ACTIONS	A.1 Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel pool water level is lower than the required level, the movement of irradiated fuel assembly in the spent fuel pool is immediately suspended. This does not preclude movement of a fuel assembly to a safe position. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.
SURVEILLANCE REQUIREMENTS	SR 3.7.15.1 This SR is done during the movement of irradiated fuel assemblies as stated in the Applicability. This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. During refueling operations, the level in the spent fuel pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.7.1.
REFERENCES	1. FSAR, Section 9.1.2. 2. FSAR, Section 9.1.3. 3. FSAR, Section 9.1.4.3.4, 15.4.5 and 15.5.22. 4. Regulatory Guide ~~1.25~~1.183, July 2000. 5. 10 CFR ~~100.11~~50.67.

B 3.7 PLANT SYSTEMS

B 3.7.18 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a ~~1~~0.75 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours). Operating at or below 0.1 μCi/gm ensures that in the event of a DBA, offsite doses will be less than 10 CFR ~~100~~50.67 requirements.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the FSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed 10 CFR ~~100~~50.67 limits (Ref. 1)~~ for whole body and thyroid dose rates~~.

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ADVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

(continued)

BASES

ACTIONS	A.1 and A.2 (continued) least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.
SURVEILLANCE REQUIREMENTS	SR 3.7.18.1 This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.
REFERENCES	1. 10 CFR ~~100.11~~50.67. 2. FSAR, Chapter 15.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources—Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources — Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of recently irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident involving handling recently irradiated fuel. Due to radioactive decay, AC electrical power is only required to mitigate fuel handling accidents involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours).

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted, provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources-Shutdown

BASES

BACKGROUND	A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources-Operating."
APPLICABLE SAFETY ANALYSES	The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation. The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY. The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of recently irradiated fuel assemblies ensures that: a. The unit can be maintained in the shutdown or refueling condition for extended periods; b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident involving handling recently irradiated fuel. Due to radioactive decay, DC electrical power is only required to mitigate fuel handling accidents involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours). The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The DC electrical power subsystems, each subsystem consisting of one battery, one battery charger per battery, and the corresponding control equipment and interconnecting class 1E cabling within the subsystem, are required to be OPERABLE to support required trains of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems-Shutdown." An OPERABLE subsystem consists of a DC bus connected to a battery with an OPERABLE battery charger which is fed from an OPERABLE AC vital bus (Ref B.3.8.10).

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters-Shutdown

BASES

BACKGROUND	A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."
APPLICABLE SAFETY ANALYSES	The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The Class 1E UPS inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protective System and Engineered Safety Features Actuation System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY. The OPERABILITY of the minimum inverters to each 120 VAC vital bus during MODES 5 and 6 and during movement of recently irradiated fuel assemblies ensures that: a. The unit can be maintained in the shutdown or refueling condition for extended periods; b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident involving handling recently irradiated fuel. Due to radioactive decay, AC and DC inverters are only required to mitigate fuel handling accidents involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours). The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling recently irradiated fuel).

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems - Shutdown

BASES

BACKGROUND

A description of the Class 1E AC, DC, and 120 VAC vital bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The Class 1E AC, DC, and 120 VAC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the Class 1E AC, DC, and 120 VAC vital bus electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum Class 1E AC, DC, and 120 VAC vital bus electrical power distribution subsystems during MODES 5 and 6, and during movement of recently irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident involving handling recently irradiated fuel. Due to radioactive decay, AC and DC electrical power is only required to mitigate fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous ~~100~~72 hours).

The Class 1E AC, DC, and 120 VAC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. An OPERABLE AC subsystem shall consist of a 4kV vital bus powered from at least one energized offsite power source with the capability of being powered from an OPERABLE DG. The DG may be the DG associated with that bus or, with administrative controls in place, a DG that can be cross-tied (via the startup cross-tie feeder breakers) to another bus. However, credit for this cross-tie capability

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

In MODES 1, 2, 3, and 4, the containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained ~~well~~ within the requirements of 10 CFR ~~100~~50.67. Additionally, in all operating modes the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions. However during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to maintain the pressure boundary can be less stringent. An analysis has been performed that shows by meeting the LCO, during CORE ALTERATION and movement of irradiated fuel assemblies in containment, the potential release as a result of a fuel handling accident (FHA) will remain ~~well~~ within the requirements of 10 CFR ~~100~~50.67 limits.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. The LCO requires that during CORE ALTERATIONS or the movement of irradiated fuel assemblies the equipment hatch must be capable of being closed and held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment Personnel Air Lock (PAL) and Emergency Air Lock (EAL), which are also part of the containment pressure boundary, provide a means for personnel and emergency access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each of these air locks has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when the PAL and EAL are not required to be closed, the door interlock mechanisms may be disabled, allowing both doors of each of the air locks to remain open for extended periods when frequent containment entry is necessary.

(continued)

BASES

BACKGROUND (continued)

Per the FHA inside containment analysis, there are no closure restrictions required to limit any release to ~~well~~ within the requirements of 10 CFR ~~100~~50.67 limits for offsite dose as the result of a fuel handling accident during refueling. The LCO requirements for containment penetration closure are not provided to meet regulatory requirements, but rather to reduce the potential volume of the release of fission product radioactivity within containment to the environment.

The Containment Purge and Exhaust System includes two subsystems. The normal subsystem includes a 48 inch purge penetration and a 48 inch exhaust penetration ~~in which the flow path is limited to being open 200 hour or less per calendar year~~. The second subsystem, a pressure equalization system provides a single 12 inch supply and exhaust penetration. ~~The three valves in the 12 inch pressure equalization penetration can be opened intermittently.~~ Each of these systems are qualified to closed automatically by the Engineered Safety Features Actuation System (ESFAS). ~~Neither of the subsystems is subject to a Specification in MODE 5.~~

In MODE 6, large air exchanges are necessary to conduct refueling operations. The normal 48 inch purge system is used for this purpose, and all four valves are closed by the ESFAS in accordance with LCO 3.3.6, "Containment Purge and Exhaust Isolation Instrumentation."

The pressure equalization system is disassembled and used in MODE 6 for other outage functions.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side if they are not opened under administrative controls. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. The fuel transfer tube is open but closure is provided by an equivalent isolation of a water loop seal. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, ventilation barrier for the other containment penetrations during fuel movements (Ref. 1).

Although the historic severe weather patterns for DCPP do not require consideration of tornados as part of the design basis, severe weather conditions might occur at the site that could necessitate closure of open penetrations with direct access to the outside atmosphere during refueling operations with core alterations or irradiated fuel movement inside containment. As a result, administrative procedures shall require that closure of these penetrations be initiated immediately if severe weather warnings are in effect. All fuel handling activities inside containment shall be suspended until closure of the equipment hatch is completed.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSIS

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accident inside the containment is based on dropping a single irradiated fuel assembly of which all 264 fuel rods rupture. In addition the analysis assumes free and rapid communication of air from the containment to the outside environment; the accident occurs ~~100~~72 hours after reactor shutdown; almost instantaneous release of the entire containment volume to the outside atmosphere; ~~thyroid dose conversion factors based on ICRP 30 (Ref. 4);~~ a radial peaking factor of 1.65 based on 105% full power operation; and the other guidance from RG ~~1.25~~1.183. (Ref 5).

The requirements of LCO 3.9.7, "Refueling Cavity Water Level," and the minimum decay time of ~~100~~72 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses ~~that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4,~~

~~Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values.~~less than the accident dose criteria specified in Table 6 of RG 1.183 (Ref. 5).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

REFERENCES

1. Design Criteria Memorandum T-16, Containment Functions.
2. FSAR, Section 15.4.5 and 15.5.22.
3. ~~NUREG-0800, Section 15.7.4, Rev. 1, July 1981~~Not Used.
4. ~~International Commission on Radiological Protection Publication 30, "Limits for Intakes of Radionuclides by Workers," 1979~~ Not Used.
5. RG ~~1.25~~1.183, July 2000.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. ~~2 and 6~~1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~≤ 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3~~the acceptance criteria of 10 CFR 50.67 (Ref. 4) and RG 1.183 (Ref. 1).

APPLICABLE SAFETY ANALYSIS

During CORE ALTERATIONS and movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide ~~1.25~~ 1.183 (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 200 (Appendix B (2) of Ref. ~~6~~1 approved in Ref. 7) to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain ~~10~~8% of I-131, 23% of I-132 and ~~10~~5% of core~~the total fuel rod iodine~~ inventory of all other iodine isotopes (Ref. ~~1~~2).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of ~~100~~72 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained ~~well~~ within allowable limits (Refs. 1and 4~~, and 5~~).

Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference ~~3~~1.

(continued)

BASES (continued)

APPLICABILITY

LCO 3.9.7 is applicable during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, and when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.15, "Fuel Storage Pool Water Level."

ACTIONS

A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. Regulatory Guide ~~1.25, March 23, 1972~~1.183, July 2000.
2. FSAR, Section 15.4.5 and 15.5.22.
3. ~~NUREG-0800, Section 15.7.4.~~ Not Used
4. 10 CFR ~~100.10~~50.67.
5. ~~Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-828, Radiological Consequences of a Fuel Handling Accident, December 1971.~~ Not Used
6. ~~Appendix B (2) of Regulatory Guide 1.183, July 2000~~Not Used.
7. License Amendment 155/155, October 21, 2002