Enclosure 5 to
ULNRC-06696

Proposed Technical Specification Bases Changes
(Mark-up – for Information Only)

(53 pages)

(continued)

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 2485 psig. 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 are hydrostatically tested at 125% (3110 psig) 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).

50.67 "Accident Source Term"

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, "Reactor Site Criteria"~~ (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and

(continued)

BASES (Continued)

50.67, "Accident Source Term,"

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, "Reactor Site Criteria,"~~ 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, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
4. 10 CFR ~~100~~.

50.67, "Accident Source Term."

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ejection of a rod also produces a time dependent redistribution of core power. Depending on initial power level, this accident is terminated by the power range neutron flux - high or low reactor trip setpoint in the FSAR analyses.

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) andthrough 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, "Reactor Site Criteria,"~~ 50.67, "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 limits are specified in the COLR.

APPLICABILITY

In MODE 2 with k_{eff} < 1.0 and in MODES 3, 4, and 5 the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits,"and LCO 3.1.6, “Control Bank Insertion Limits.”

Since this Specification has no LCO 3.0.4.c allowance, MODE 5 can not be entered from MODE 6 while not meeting the SDM limits. This assures that the initial condition assumptions of an inadvertent boron dilution event in MODE 5 are met. The risk assessments of LCO 3.0.4.b may only be utilized for systems and components, not Criterion 2 values or parameters such as SDM. Therefore, a risk assessment per LCO 3.0.4.b to allow MODE changes with single or multiple system/equipment inoperabilities may not be used to allow a MODE change into or ascending within this LCO while not meeting the SDM limits, even if the risk assessment specifically includes consideration of SDM.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

In the event that a rod is known to be untrippable and not fully inserted, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth.

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

REFERENCES

50.67, "Accident Source Term."

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Chapter 15, Section 15.1.5.
3. FSAR, Chapter 15, Section 15.4.6.
4. 10 CFR ~~100~~.
5. Westinghouse NSAL-02-14.

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, defined in this specification as the Allowable Values except for Trip Functions 14.a and 14.b in Technical Specification Table 3.3.1-1 (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions), 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 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 DNBR limit;

and control room doses

2. Fuel centerline melt shall not occur; and

3. The RCS pressure Safety Limit (SL) of 2735 psig shall not be exceeded.

50.67

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~~ criteria during AOOs.

and control room

s

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 off~~site~~ dose shall be maintained within an acceptable fraction of 10 CFR ~~100 limits~~. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence.

50.67 limits per Regulatory Guide 1.183.

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

e. Steam Line Pressure - Low (continued)

This Function is anticipatory in nature and has a lead/lag ratio of 50/5.

Steam Line Pressure - Low must be OPERABLE in MODES 1, 2, and 3 (above P-11 and below P-11 unless the Safety Injection - Steam Line Pressure - Low Function is blocked) when a secondary side break or stuck openvalve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, feed line break is not a concern. Inside containment SLB will be terminated by automatic SI actuation via Containment Pressure - High 1, and outside containment SLB will be terminated by the Steam Line Pressure - Negative Rate - High signal for steam line isolation. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to have a significant effect on required plant equipment.

2. Containment Spray

Containment Spray provides three primary functions:

1. Lowers containment pressure and temperature after an HELB in containment;

2. Reduces the amount of radioactive iodine ~~in~~ the containment atmosphere; and

and particulates

3. Adjusts the pH of the water in the containment recirculation sumps after a large break LOCA, in conjunction with the Recirculation Fluid pH Control (RFPC) system.

These functions are necessary to:

- Ensure the pressure boundary integrity of the containment structure;

- Limit the release of radioactive iodine to the environment in the event of a failure of the containment structure; and

and particulates

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

In the postulated fuel handling accident, the dose calculations performed
and control room
nce 5 (for allowing the personnel airlock to be open during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment) do not assume automatic containment purge isolation. (See also the Bases for LCO 3.9.4, "Containment Penetrations.") Containment isolation 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~~ (Ref. 1) limits.

Containment mini-purge isolation 50.67 and Regulatory Guide 1.183 (isolation within 11 seconds) in the

and Ref. 8

(Reference 6). Automatic isolation of the containment mini-purge valves is also assumed in the minimum containment pressure analysis for ECCS performance capability, as described in the FSAR (Reference 7). The effect of having the containment mini-purge system in operation at the onset of the most limiting case (i.e., a double-ended cold leg guillotine break), followed by assumed automatic isolation of the containment purge exhaust and supply lines, is addressed in the analysis ~~by increasing the assumed containment volume. Consequently, the response time (11 seconds) assumed for the automatic isolation of the mini-purge system determines the adjustment made to the containment volume in the analysis~~.

in Reference 6

The containment purge isolation instrumentation satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

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

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate Containment Purge Isolation at any time by using either of two push buttons in the control room.

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 3.3.6.4 (continued)

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The channels tested have no setpoints associated with them.

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

SR 3.3.6.5

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.

SR 3.3.6.6

SR 3.3.6.6 is the performance of the required response time verification on those functions with time limits provided in Reference 3. The Surveillance Frequency is based on operating experience, equipmentreliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR ~~100.11~~ 50.67.
2. NUREG-1366, July 22, 1993.
3. FSAR Table 16.3-2.
4. Callaway OL Amendment No. [illegible]
5. Callaway OL Amendment No. [illegible]
6. FSAR Section 15.6.5.4.1.4.
7. FSAR Section 6.2.1.5.
8. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

B 3.3 INSTRUMENTATION

B 3.3.8 Emergency Exhaust System (EES) Actuation Instrumentation

BASES

BACKGROUND

The EES ensures that radioactive materials in thefuel building atmosphere following a fuel handling accident are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.13, "Emergency Exhaust System." The system initiates filtered exhaust from the fuel building following receipt of a fuel building ventilation isolation signal (FBVIS), initiated manually or automatically upon a high radiation signal (gaseous).

High gaseous radiation, monitored by two channels, provides an FBVIS. Both EES trains are initiated by high radiation detected by either channel. Each channel contains a gaseous monitor. High radiation detected by either monitor initiates fuel building isolation, starts the EES, and initiates a CRVIS. These actions function to prevent exfiltration of contaminated air by initiating filtered exhaust, which imposes a negative pressure on the fuel building. Since the radiation monitors include an air sampling system, various components such as sample line valves and sample pumps are required to support monitor OPERABILITY. In the FBVIS mode, each train is capable of maintaining the fuel building at a negative pressure of less than or equal to 0.25 inches water gauge relative tothe

50.67 (Ref. 2) and Regulatory Guide 1.183 (Ref. 5)

The EES is also actuated in the LOCA (SIS) mode as described in the Bases for LCO 3.3.2, "ESFAS Instrumentation."

APPLICABLE SAFETY ANALYSES

The EES ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident are filtered and adsorbed prior to being exhausted to the environment as discussed in Reference 1. 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. 2)~~ and control room habitability is maintained.

and control room

The EES actuation instrumentation satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).

In the fuel handling accident analysis, the EES is credited as the release point, but no credit is taken for filtration of the release.

LCO

The LCO re ... ate
the EES is ...

(continued)

BASES (Continued)

50.67

REFERENCES

1. FSAR Section 15.7.4.

2. 10 CFR ~~100.11~~.

3. FSAR Section 7.3.3 and Table 7.3-5.

4. FSAR Table 16.3-2.

5. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

BASES (Continued)

through

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address RCS Operational LEAKAGE. However, the other forms of RCS Operational LEAKAGE are related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analyses for events resulting in steam discharge to the atmosphere assume that primary to secondary LEAKAGE ~~throught~~ all steam generators (SGs) is one gallon per minute. To a lesser extent, other it 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 safety analyses.

activity

primary to secondary

Primary to secondary LEAKAGE is a factor in the ~~dose~~ releases outside containment resulting from a steam line break (SLB) accident. ~~Other~~ accidents or transients involving secondary steam release to the atmosphere include the steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

that some of the

The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released via a postulated stuck-open atmospheric steam dump (ASD) valve or a partially stuck-open main steam safety valve. The 1 gpm primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential for SGTR given the magnitude of the postulated break flow rate.

involving secondary steam release to the atmosphere

The safety analysis for the SLB accident assumes the entire 1 gpm primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB and ~~SGTR~~ accidents are ~~well~~ within the limits defined in 10 CFR ~~100 (Ref. 5) (i.e., a 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).

The RCS operational LEAKAGE satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

50.67 (Ref. 5) and Regulatory Guide 1.183 (Ref. 9)

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure BoundaryLEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued

(continued)

BASES (Continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 4 and 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 15.6.3.
4. NUREG-1061, Volume 3, November 1984.
5. 10 CFR ~~100~~ 50.67.
6. Amendment No. 116 dated October 1, 1996.
7. NEI 97-06, "Steam Generator Program Guidelines."
8. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."
9. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

for any 2-hour time period

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 (Ref. 1)~~. Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room 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. T 50.67 (Ref 1) and Regulatory minimize the dose consequences in th Guide 1.183 (Ref. 2). or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for bot 1 and Ref. VALENT 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~~ (Ref. 2).

10 CFR 50.67 and Regulatory Guide 1.183

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following an SLB and SGTR accident. The safety analyses (Refs. 3 and 4) assume the initial iodine specific activity of the reactor coolant is greater than the LCO limit (see the discussion of Case 1 below), and a pre-accident reactor coolant steam generator (SG) tube leakage rate of 1 gpm exists. The safety analyses assume the initial ~~iodine~~ specific activity of the secondary coolant is ~~10% of the Case 1 reactor coolant iodine specific activity, greater than~~ the limit of 0.1 μCi/gm DOSE EQUIVALENT I -131 from LCO 3.7.18, "Secondary Specific Activity."

iodine

at

The analyses 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 plant that could affect RCS specific activity, as they relate to the acceptance limits.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.16.2 (continued)

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.

REFERENCES

1. 10 CFR ~~100.11, 1973~~ 50.67.

2. ~~Standard Review Plan (SRP), Section 15.1-5 Appendix A (SLB) and Section 15.6.3 (SGTR)~~ Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

3. FSAR, Section 15.1.5.

4. FSAR, Section 15.6.3.

BASES (Continued)

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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 that some of assumes a primary to secondary LEAKAGE rate of 1 gpm to the unaffected steam generators, exceeding the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via a postulated stuck-open atmospheric steam dump (ASD) valve or via a partially stuck-open main steam safety valve (see Ref. 2).

50.67

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). In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute. 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. 3), 10 CFR ~~100~~ (Ref. 4) ~~or the NRC approved licensing basis (e.g., a small fraction of theselimits)~~.

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

, and Regulatory Guide 1.183 (Ref. 8)

LCO

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

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

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.17.2 (continued)

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 plugging 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 plugging 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. FSAR Section 15.6.3.
3. 10 CFR 50 Appendix A, GDC 19.
4. 10 CFR ~~100~~. 50.67.
5. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
6. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
7. EPRI TR-107569, "Pressurized Water Reactor Steam Generator Examination Guidelines."
8. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

BASES

BACKGROUND (continued)

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

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

c. All equipment hatches are closed and sealed; and

d. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

APPLICABLE SAFETY ANALYSES

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

and control room

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA) and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.2% of containment air weight per day for the first 24 hours and 0.1% of containment air weight per day for the remainder of the accident (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design bases LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.2% of containment air weight per day at P_a = 48.1 psig. This is a conservative value for P_a since the calculated peak containment pressure for LOCA is 47.8 psig.

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

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

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met.

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

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

The personnel air lock is nominally a right circular cylinder, approximately 10 ft in diameter, with a door at each end. The emergency air lock is approximately 5 ft 9 in inside diameter with a 2 ft 6 in door at each end. On both air locks, doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock is provided with limit switches on both doors that provide local indication of door position. and rod ejection accident

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the safety analyses.

are s these accidents

APPLICABLE SAFETY ANALYSES

The DBA that result in a release of radioactive material within containment ~~is~~ a loss of coolant accident (Ref. 2). In the analysis of ~~this accident~~, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.2% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B, (Ref. 1) ~~L_a~~ as the maximum allowable containment leakage rate at the calculated peak containment internal pressure, P_a = 48.1 psig following a design basis LOCA (see the Applicable Safety Analysis Bases for LCO 3.6.1, "Containment."). This

L_a, (continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)

supports le Closure of the 18 inch mini-purge valves within analyses
of any even 11 seconds is incorporated in the large break is LCO.
LOCA dose calculation.

The DBAs t
containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 1). In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment shutdown purge and mini-purge valves) are minimized. The safety analyses assume that the 36 inch Containment Shutdown Purge ~~and 18 inch Mini-Purge~~ valves are closed at event ~~initiation,~~ initiation; however, the penetration flow paths may be isolated by blind flanges.

The DBA analysis assumes that isolation of the containment is complete and leakage terminated except for the design leakage rate, L_a.

The ~~LOCA offsite dose analysis assumes~~ LOCA and rod ejection analyses assume leakage from the containment at a maximum leak rate of 0.20 percent of the containment air weight per day for the first 24 hours, and at 0.10 percent of the containment air weight 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 spring-closed of the 18 inch containment mini-purge valves. ~~Two valves in series on each~~ purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves are pneumatically operated ~~spring closed~~ valves that will fail closed on the loss of air.

The 36 inch Containment Shutdown Purge and exhaust valves may be unable to close against containment pressure following a LOCA. Therefore, either each of the Containment Shutdown Purge and exhaust valves is required to remain sealed closed during MODES 1, 2, 3, and 4 or closed and blind flanges must be installed. The Containment Shutdown 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 10 CFR 50.36(c)(2)(ii).

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray and Cooling Systems

BASES

fission product

BACKGROUND

The Containment Spray and Containment Cooling system provides containment atmosphere cooling to limit post accident pressure and temperature (see B 3.6.5) in containment to less than the design values. Reduction of containment pressure and the ~~iodine~~ removal and retention capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray and Containment Cooling system are designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," GDC 43, "Testing of Containment Atmosphere Cleanup Systems" and GDC 50, "Containment Design Basis" (Ref. 1).

The Containment Cooling System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Containment Spray System and the Containment Cooling System provide complementary methods to limit and maintain post accident conditions to less than the containment design values.

Containment Spray System

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment sumps.

The Containment Spray System provides a spray of borated water mixed with trisodium phosphate from the Recirculation Fluid pH Control baskets 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 temperature and pressure reducing

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 2).

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a -2.98 psig containment pressure and is associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-3 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), sequenced loading of equipment, containment spray pump startup, and spray line filling (Ref. 4).

Containment cooling train performance for post accident conditions is given in Reference 4. The result of the analysis is that each train can provide 100% of the required peak cooling capacity during the post accident condition. The train post accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference 4.

The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-3 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 Essential Service Water pump startup times.

The Containment Spray System and the Containment Cooling System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, a minimum of one containment cooling train and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits (Refs. 3). Additionally, one containment spray train is also required to remove ~~iodine~~ fission products from the

(continued)

BASES

LCO (continued)

containment atmosphere and retain volatile iodine species in the sumps, consistent with the safety analysis. To ensure that these requirements are met, two containment spray trains and two containment cooling trains must be OPERABLE. Therefore, in the event of an accident, at least one train in each system operates, assuming the worst case single active failure occurs.

A 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 and manually transferring to the containment sump. In addition, management of gas voids is important to Containment Spray System OPERABILITY. The containment Spray System is OPERABLE when it is sufficiently filled with water to perform its specified safety function.

A Containment Cooling train typically includes cooling coils, dampers, two 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 containment coolingtrains.

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.

ACTIONS

A.1

fission product

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the ~~iodine~~ removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant temperature and pressure reducing capability afforded by the Containment Spray System, reasonable time for repairs, and low probability of a DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low

(continued)

BASES

ACTIONS

A.1 (continued)

probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time.

Refer to Section 1.3, "Completion Times," for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

B.1 and B.2

If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time, 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 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time for attempting restoration of the containment spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

fission product

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

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

(continued)

BASES (Continued)

APPLICABLE SAFETY ANALYSES

The RFPC System is essential to the removal and retention of airborne iodine within containment following a DBA.

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value volume following the accident. The analysis assumes that more than 90% of containment is covered by the spray (Ref. 1).

The DBA response time assumed for the RFPC System is the same as for the Containment Spray System and is discussed in the Bases for LCO 3.6.6, "Containment Spray and Cooling Systems."

The DBA analyses assume that one train of the Containment Spray System is inoperable.

The RFPC System satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The RFPC System is necessary to reduce the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume of TSP-C must be sufficient to raise the average long term containment sump solution pH to a level conducive to iodine removal and retention, namely, to greater than 7.1. This pH level maximizes the effectiveness of the iodine removal and retention mechanisms without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the RFPC System. The RFPC System assists in reducing the iodine fission product inventory prior to release to the environment.

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

ACTIONS

A.1

If the RFPC System is inoperable, it must be restored to OPERABLE within 72 hours. The pH adjustment of the Containment Spray System flow for corrosion protection and iodine removal/retention enhancement is reduced in this condition. The Containment Spray System would still be available and would remove some ~~iodine~~ fission products from the containment

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

MSLPDIVs isolates the break and limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs, MSIVBVs, and MSLPDIVs.

d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiologicalreleases.

e. The MSIVs, MSIVBVs, and MSLPDIVs are also utilized during other less limiting events such as a feedwater line break.

Figure B 3.7.2-1 is a curve of the MSIV isolation time as function of steam generator pressure. Meeting the MSIV isolation times in Figure B 3.7.2-1 ensures that the evaluation performed in Reference 7 remains valid.

The MSIVs, MSIVBVs, and MSLPDIVs satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

This LCO requires that the MSIV and its associated actuator trains, the MSIVBV, and the MSLPDIV for each of the four main steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within the limits of Figure B 3.7.2-1 and they are capable of closing on an isolation actuation signal. An MSIV actuator train is considered OPERABLE when it is capable of closing its associated MSIV on an isolation actuation signal. The MSIVBVs and MSLPDIVsare considered OPERAB and control room n times are withinlimits and they are capable of closing on an isolation actuation signal.

This LCO provides assurance that the MSIVs, MSIVBVs, and MSLPDIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR ~~100~~ (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs, MSIVBVs, and MSLPDIVs must be OPERABLE in MODES 1, 2 and 3, when there is significant mass and energy in the RCS and steam generators.

or Regulatory Guide 1.183 (Ref. 8)

50.67

Steam is supplied to the turbine and other loads from the four steam generators by four main steam lines. One MSIV and MSIVBV is installed in each of the four main steam lines. One MSLPDIV is installed in the drain line off each of the four main steam lines. When the main steam line

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.3 (continued)

may also be required to be performed upon returning the unit to operation following a refueling outage.

The Frequency for this SR is in accordance with the INSERVICE TESTING PROGRAM.

REFERENCES

1. FSAR, Section 10.3, Main Steam Supply System.

2. FSAR, Section 6.2, Containment Systems.

3. FSAR, Section 15.1.5, Steam System Piping Failure.

4. 10 CFR ~~100.11~~. 50.67.

5. FSAR 6.2.1.4.3.3, Containment Pressure-Temperature Results.

6. Amendment 172 to Facility Operating License No. NPF-30, (NRC Safety Evaluation included), Callaway Unit 1, dated June 16, 2006.

7. Westinghouse Letter, SCP-07-26, dated March 6, 2007.

8. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

all available

APPLICABLE SAFETY ANALYSES (continued)

In the accident analysis presented in Reference 2, the ASDs are assumed to be used by the operator to cool down the unit to RHR entry conditions for accidents accompanied by a loss of offsite power. The main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. The time associated with unit cooldown to RHR entry conditions for these accident scenarios determines the required duration of ASD operation.

With a single failure of the one ASD

For the recovery from a steam generator tube rupture (SGTR) event described in Reference 3, the operator is also required to perform a rapid cooldown using ~~two~~ intact steam generators to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to RHR conditions for this event and also for other accidents. Thus, the SGTR is the limiting event for ASD performance. The number of ASDs required to be OPERABLE to satisfy the SGTR accident analysis requirements is four. ~~This accounts for the unavailable (isolated) ASD~~ associated with the ruptured SG, ~~thereby leaving two ASDs available~~ for heat removal.

three ASDs would remain OPERABLE

The ASDs are equipped with manual isolation valves in the event an ASD spuriously fails open or fails to close during use.

The ASDs satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Four ASD lines are required to be OPERABLE. One ASD line is required from each of four steam generators to ensure that at least two intact SG ASD lines are available to conduct the rapid RCS cooldown following an SGTR, in which one steam generator becomes unavailable due to the steam generator tube rupture, accompanied by a single, active failure of a second ASD line on an unaffected steam generator. The manual isolation valves must be OPERABLE to isolate a failed open ASD line. The accident analyses that credit OPERABILITY of the ASDs require them to relieve steam to the atmosphere in order to perform their safety related function.

Failure to meet the LCO can result in the inability to achieve subcooling, consistent with the assumptions used in the steam generator tube rupture analysis, to facilitate equalizing pressures between the reactor coolant system and the ruptured steam generator.

An ASD is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and is capable of fully opening

(continued)

B 3.7 PLANT SYSTE[illegible]

B 3.7.10 Control Roo[illegible]

Although each pressurization system train includes a charcoal adsorber section (and associated heater), the charcoal adsorber is not credited in the Alternative Source Term (AST) radiological consequence analyses.

BASES

BACKGROUND

The CREVS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity. The CREVS consists of two independent, redundant trains that pressurize, recirculate, and filter the control room air. Each CREVS train consists of a filtration system train and a pressurization system train. Each filtration system train consists of a fan, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a second HEPA filter follows the adsorber section to collect carbon fines. Each pres[illegible]ion system train consists of a fan, a moisture separator, an electric heater, a HEPA filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a second HEPA filter follows the adsorber section to collect carbon fines. Ductwork, valves or dampers, and instrumentation also form part of the CREVS system.

that

The CREVS is an emergency system which may also operate during normal unit operations. Upon receipt of the actuating signal, normal air supply and exhaust to the control room envelope (CRE) is isolated, a [illegible] system filter trains, [illegible] remove any large [illegible]ny entrained water [illegible]EPA filters and charcoal adsorbers. Continuous operation of each pressurization system train for at least 15 minutes per month, with the heaters functioning, reduces moisture buildup on the HEPA filters and adsorbers. The heaters are important to the effectiveness of the charcoal adsorbers.

;however, the charcoal adsorbers in the pressurization system train are not credited in the AST radiological consequence analyses

Actuation of the CREVS by a Control Room Ventilation Isolation Signal (CRVIS), places the system in the emergency mode of operation. Actuation of the system to the emergency mode of operation closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the air within the CRE through the redundant trains of HEPA and the charcoal filters. The emergency (CRVIS) mode also initiates pressurization and filtered ventilation of the air supply to the CRE.

The control room pressurization system draws in outside air, processing it through a particulate filter charcoal adsorber train for cleanup. This outside air is diluted with air drawn from the cable spreading rooms and the electrical equipment floor levels within the control building and distributed back into those spaces for further dilution. The control room filtration units take a portion of air from the exhaust side of the

(continued)

downstream

BASES

BACKGROUND (continued)

system, ~~upstream~~ of the outside air ~~intake, for dilution with portions of the exhaust air from the control room air-conditioning system and processes it~~ through the control room filtration system adsorption train for additional

intake. The control room air conditioning system (CRACS) supplies both the CRE and the Equipment Room Envelope (ERE) with forced air flow. After dilution in the ERE, a portion of the control room air conditioning system discharge is mixed with pressurization flow which is then processed

ther diluted with the remaining control room um air, cooled, and supplied to the CRE. This R E under a positive envelope (CBE) water o the outside atmosphere. This will assure hus preventing any unprocessed contaminants

building pressure boundary during normal operation is continuously monitored by radiation, carbon dioxide/ monoxide, and smoke detectors. A high radiation signal initiates the emergency (CRVIS) mode of operation; the other detectors provide an

In addition to the CBE, airflow from the ERE is mixed at a nominal 300 cfm as part of the 2000 cfm filtration flow. The balance of 1300 cfm (nominal) flow through the filter is supplied by pressurized CRE outflow that is supplied to the intake of the control room filtration system.

ation monitors lation Signal, haust radiation nstrumentation associated with actuation of the CREVS is addressed in LCO 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation."

A single CREVS train will pressurize the CRE to about 0.125 inches water gauge relative to the outside environment. The 0.125 inches water gauge positive pressure is obtained based on a nominal flowrate of 2000 cfm through the filtration filter, which includes 400 cfm of control building envelope (CBE) air. The CREVS operation in maintaining the control room habitable is discussed in the FSAR, Section 6.4 and 9.4 (Ref. 1 and 9).

Redundant pressurization and filtration trains provide the required filtration should an excessive pressure drop develop across the other filter. 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 CREVS is designed in accordance with Seismic Category I requirements.

dose of 5 rem TEDE (Ref. 11).

The CREVS is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a ~~5 rem whole body dose or its equivalent to any part of the body~~.

By operation of the control room pressurization trains and the control room filtration units, the CREVS pressurizes, recirculates and filters air within the CRE as well as the CBE that generally surrounds the CRE. The boundaries of these ~~two~~ distinct but related volumes are credited in the analysis of record for limiting the inleakage of unfiltered outside air.

three

(continued)

,

,and the ERE which is adjacent to the CRE

BASES

BACKGROUND (continued)

The plant CRE design is unique. The Control Building by and large surrounds the CRE. The Control Building is also designed to be at a positive pressure with respect to its surrounding environment although not positive with respect to the CRE. In the emergency pressurization and filtration mode, the control room air volume receives air through a filtration system that takes suction on the Control Building. The Control Building ~~is~~ in turn receives filtered air from the outside environment.

The ERE is an analytically separate subvolume within the CBE. This subvolume encloses the CRACS and control room filter unit (CRFU), among other equipment. Occupancy of the ERE is not required to control the unit during normal and accident conditions. The ERE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the ERE. The ERE boundary must be maintained to ensure that the inleakage of unfiltered air into the ERE will not exceed the inleakage assumed in the licensing basis analysis of the design basis event consequences to CRE occupants. The ERE and its boundary are defined in the Control Room Envelope Habitability Program.

the CRE. The CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The CBE is an area that largely surrounds the CRE. Occupancy of the CBE is not required to control the unit during normal and accident conditions. The CBE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CBE. The CBE boundary must be maintained to ensure that the inleakage of unfiltered air into the CBE will not exceed the inleakage assumed in the licensing basis analysis of DBA consequences to CRE occupants. The CBE and its boundary are defined in the Control Room Envelope Habitability Program.

APPLICABLE SAFETY ANALYSES

The CREVS 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 CREVS 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 15A.3 (Ref. 2).

The worst case single active failure of a component of the CREVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The CREVS provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases (Ref. 7) determined that hazardous chemicals are not stored or used onsite in quantities sufficient to necessitate CRE protection as required by Regulatory Guide 1.78 (Ref. 8). The evaluation of a smoke challenge demonstrates that such an event 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 analysis for smoke and hazardous chemical releases accordingly assumes no CREVS actuation for such events.

The CREVS satisfies Criterion 3 of 10 CFR 50.3 , CBE or ERE

LCO

Two independent and redundant CREVS trains are required to be OPERABLE to ensure that at least one is available if a single active failure disables the other train TEDE system failure, such as from a loss of both ventilation trains or from an inoperable CRE ~~or CBE~~ boundary, could result in exceeding a dose of 5 rem ~~whole body or its equivalent to any part of the body~~ to the CRE occupants in the event of a large radioactive release.

(except that a charcoal adsorber is not required for either of the pressurization system trains) idual components necessary to limit CRE occupants exposure are OPERABLE. A CREVS train is OPERABLE when the associated:

(except that a heater is not required for either of the pressurization system trains) n fans are

, CBE and ERE

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

c. Heater, moisture separator, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In order for the CREVS trains to be considered OPERABLE, the CRE ~~and CBE~~ boundaries must be maintained such that the CRE occupant dose from a large radioactive release doe , CBE and ERE lated dose in the licensing basis consequence analyses for DBA's.

The LCO is modified by a Note allowing the CRE ~~and CBE~~ boundaries to be opened intermittently under administrative controls. This Note only applies to openings in the CRE ~~and CBE~~ boundaries that can be rapidly restored to the intended design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings these controls should be

, CBE or ERE

(continued)

BASES

LCO (continued)

procedurализed and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and thereby restore the affected envelope boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

Note that the Control Room Air Conditioning System (CRACS) forms a subsystem to the CREVS. The CREVS remains capable of performing its safety function provided the CRACS air flow path is intact and air circulation can be maintained. Isolation or breach of the CRACS air flow path can also render the CREVS flow path inoperable. In these situations, LCOs 3.7.10 and 3.7.11 may be applicable.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREVS must be OPERABLE to ensure that the CRE will remain habitable during and following a LOCA or SGTR.

During movement of irradiated fuel assemblies, the CREVS must be OPERABLE to cope with the release from a design basis fuel handling accident , CBE or ERE r in the fuel building.

ACTIONS

A.1

When one CREVS train is inoperable for reasons other than an inoperable CRE ~~or CBE~~ boundary, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a failure in the OPERABLE CREVS train could result in loss of CREVS 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

If the unfiltered inleakage of potentially contaminated air past a CRE or CBE boundary credited in the accident analysis and into the CRE 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), actions must be taken to restore the affected boundary (or boundaries) to OPERABLE status within 90 days.

(continued)

, CBE or ERE

ACTIONS

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

During the period that a CRE ~~or CBE~~ boundary credited in the accident analysis is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological event. 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. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE or CBE boundary of an envelope credited in the accident analysis) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional.

The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan, and possibly repair and test most conditions adversely affecting the CRE ~~or CBE~~ boundary credited in the accident analysis.

, CBE or ERE

C.1 and C.2

In MODE 1, 2, 3, or 4, if the inoperable CREVS train or inoperable CRE or CBE boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1, D.2.1, and D.2.2

During movement of irradiated fuel assemblies, if the inoperable CREVS train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREVS train in the CRVIS mode. This action ensures

ACTIONS

D.1, D.2.1, and D.2.2 (continued)

that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. Required Actions D.2.1 and D.2.2 would place the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

E.1 and E.2

During movement of irradiated fuel assemblies, with two CREVS trains inoperable or one or more CREVS trains inoperable due to an inoperable CRE ~~or CBE~~ boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

, CBE or ERE

F.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4, for reasons other than an inoperable CRE ~~and CBE~~ boundary (i.e., Condition B), the CREVS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

, CBE or ERE

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train periodically, by initiating from the control room, flow through the HEPA filters and charcoal adsorbers of both the filtration and pressurization systems, provides an adequate check of this system.

Periodic heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. Each pressurization system

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1 (continued)

but they may

train must be operated for ≥ 15 continuous minutes ~~with the heaters functioning~~. Functioning heaters will not necessarily have the heating elements energized continuously for 15 minutes; ~~but will~~ cycle depending on the air temperature. Each filtration system train need only be operated for ≥ 15 minutes continuously to demonstrate the function of the system. The 15-minute run time is based on Position C.6.1 of Reference 10. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

SR 3.7.10.2

This SR verifies that the required CREVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP).

The CREVS filter tests use the test procedure guidance in Regulatory Guide 1.52 (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 3.7.10.3

This SR verifies that each CREVS train starts and operates on an actual or simulated actuation signal. The actuation signal includes Control Room Ventilation Isolation or Fuel Building Ventilation Isolation. The CREVS train automatically switches on an actual or simulated CRVIS signal into a CRVIS mode of operation with flow through the HEPA filters and charcoal adsorber banks. The Surveillance Requirement also verifies that a control room ventilation isolation signal (CRVIS) will be received by the LOCA sequencer to enable an automatic start of the Diesel Generator loads that are associated with a CRVIS. Verification that these loads will start and operate at the appropriate step in the LOCA sequencer and that other auto-start signals for these loads will be inhibited until the LOCA sequencer is reset is accomplished under Surveillance Requirement SR 3.8.1.12. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

, CBE or ERE

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.4

This SR verifies the OPERABILITY of the CRE ~~and CBE~~ boundaries credited in the accident analysis by testing for unfiltered air inleakage past the credited envelope boundaries and into the CRE. The details ofthe testing are specified in the Control Room Envelope Habitability Program.

TEDE

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~~ and the CRE occupants are protected from hazardous chemicals and smoke. For Callaway, there is no CREVS actuation for hazardous chemical releases or smoke and there are no Surveillance Requirements that verify OPERABILITY for hazardous chemicals or smoke. This SR verifies that the unfiltered air inleakage into CRE and CBE boundaries credited in the accident an an the flow rate assumed in the licensing basis ana ue nces.

start a new paragraph

start a new paragraph

When unfiltered air inleakage is greater tha te, Condition B must be entered. Required Action B.3 allows time to restore the envelope boundary credited in the accident analysis to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 4) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref 5). These compensatory measures may also be used as mitigating actions as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures to ~~restore OPERABILITY~~ (Ref. 6). Options for restoring the envelope boundary credited in the accident analysis to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the envelope boundary credited in the accident analysis, or a combination of these actions. Depending upon the nature akage test may no credited in the acc

support OPERABILITY determinations for addressing degraded/non-conforming conditions associated with the CRE and control room habitability

REFERENCES

1. FSAR, Section 6.4, Habitability Systems.

2. FSAR, Chapter 15A.3, Control Room Radiological Consequences Calculation Models.

3. Regulatory Guide 1.52, Rev. 2, Design, Testing, and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants.

(continued)

REFERENCES (continued)

4. Regulatory Guide 1.196, “Control Room Habitability at Light-Water Nuclear Power Reactors,” Revision 1.

5. NEI 99-03, “Control Room Habitability Assessment,” June 2001.

6. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, “NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability.” (ADAMS Accession No. ML040300694).

7. FSAR Section 2.2, Nearby Industrial, Transportation, and Military Facillities.

8. Regulatory Guide 1.78, “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release,” Rev. 0.

9. FSAR Section 9.4, Air Conditioning, Heating, Cooling, and Ventilation.

10. Regulatory Guide 1.52 (Rev. 3), Design, Inspection and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants.

11. 10 CFR50, Appendix A, GDC 19.

B 3.7 PLANT SYSTEMS

B 3.7.13 Emergency Exhaust System (EES)

BASES

BACKGROUND

The Emergency Exhaust System serves both the auxiliary building and the fuel building. Following a safety injection signal (SIS), safety related dampers isolate the auxiliary building, and the Emergency Exhaust System exhausts potentially contaminated air due to leakage from ECCS systems. The Emergency Exhaust System also can filter airborne radioactive particulates from the area of the fuel pool following a fuel handling accident.

The Emergency Exhaust System consists of two independent and redundant trains. Each train consists of a heater, a prefilter, a high efficiency particulate air (HEPA) filter bank, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, dampers, and instrumentation also form part of the system. A second bank of HEPA filters follows the adsorber section to collect carbon fines.

The Emergency Exhaust System is on standby for an automatic start following receipt of a fuel building ventilation isolation signal (FBVIS) or a safety injection signal (SIS). Initiation of the SIS mode of operation takes precedence over any other mode of operation. In the SIS mode, the system is aligned to exhaust the auxiliary building. The instrumentation associated with actuation of the SIS mode of operation is addressed in LCO 3.3.2, ESFAS Instrumentation.

Upon receipt of a fuel building ventilation isolation signal generated by gaseous radioactivity monitors in the fuel building exhaust line, normal air discharges from the building are terminated, the fuel building is isolated, the stream of ventilation air discharges through the system filter trains, and a control room ventilation isolation signal (CRVIS) is generated. The instrumentation associated with actuation of the FBVIS mode of operation is addressed in LCO 3.3.8, EES Actuation Instrumentation.

The Emergency Exhaust System is discussed in the FSAR, Sections 6.5.1, 9.4.2, 9.4.3, and 15.7.4 (Refs. 1, 2, 3 ~~and 4~~ respectively) because it may be used for normal, as well as post accident, atmospheric cleanup functions.

,15.6.5.4.1.2 ,4 and 5,

(continued)

(Ref. 5) credits the EES for the release point but no credit is taken for filtration of the release.

(Ref. 4)

APPLICABLE SAFETY ANALYSES

The Emergency Exhaust System design basis is established by the consequences of two Design Basis Accidents (DBAs), which are a loss of coolant accident (LOCA) and a fuel handling accident (FHA). ~~The analysis of the fuel handling accident, given in Reference 4, assumes that all fuel rods in an assembly are damaged.~~ The analysis of the LOCA assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) and Containment Spray System during the recirculation mode are filtered and adsorbed by the Emergency Exhaust System. The DBA analysis of the fuel handling accident ~~and of the LOCA assumes that only one train of the Emergency Exhaust System is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the fuel building is determined for a fuel handling accident and for a LOCA. These~~ assumptions and the analysis follow the guidance provided in Regulatory ~~Guides 1.4 (Ref. 6) and 1.25 (Ref. 5)~~.

Guide 1.183 (Ref. 6).

The Emergency Exhaust System satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Two independent and redundant trains of the Emergency Exhaust System are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the othertrain. Total system failure could result in the atmospheric release from the auxiliary building or fuel building exceeding regulatory release limits in the event of a LOCA or fuel handling accident.

In MODES 1, 2, 3 and 4 the Emergency Exhaust System (EES) is considered OPERABLE when the individual components necessary to control releases from the auxiliary building are OPERABLE in both trains (i.e., the components required for the SIS mode of operation and the auxiliary building pressure boundary). During movement of irradiated fuel assemblies in the fuel building, the EES is considered OPERABLE when the individual components necessary to control releases from the fuel building are OPERABLE in both trains (i.e. the components required for the FBVIS mode of operation and the fuel building pressure boundary). An Emergency Exhaust System train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.4

This SR verifies the integrity of the auxiliary building enclosure. The ability of the auxiliary building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the Emergency Exhaust System. During the SIS mode of operation, the Emergency Exhaust System is designed to maintain a slight negative pressure in the auxiliary building, to prevent unfiltered leakage. The Emergency Exhaust System is designed to maintain a negative pressure ≥ 0.25 inches water gauge with respect to atmospheric pressure at the flow rate specified in the VFTP. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

15.6.5.4.1.2, Radioactive Release Due to Leakage from ECCS and Containment Spray Recirculation Lines.

This SR verifies the integrity of the fuel building enclosure. The ability of the fuel building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the Emergency Exhaust System. During the FBVIS mode of operation, the Emergency Exhaust System is designed to maintain a slight negative pressure in the fuel building, to prevent unfiltered leakage. The Emergency Exhaust System is designed to maintain a negative pressure ≥ 0.25 inches water gauge with respect to atmospheric pressure at the flow rate specified in the VFTP. 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 6.5.1, Engineered Safety Features (ESF) Filter Systems.

2. FSAR, Section 9.4.2, Fuel Building HVAC.

FSAR, Section 15.7.4, Fuel Handling Accidents.

Section 9.4.3, Auxiliary Building HVAC.

4. FSAR, Section ~~15.7.4, Fuel Handling Accidents~~.

5. ~~Regulatory Guide 1.25, Rev. 0, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors~~.

(continued)

BASES

REFERENCES (continued)

6. Regulatory Guide ~~1.4, Rev. 2, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident from Pressurized WaterReactors~~.

7. Regulatory Guide 1.52 (Rev. 2), Design, Testing and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants.

8. NUREG-0800, Section 6.5.1, Rev. 2, July 1981, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.

9. Procedure EDP-ZZ-04107, HVAC Pressure Boundary and Watertight DoorControl.

10. Regulatory Guide 1.52 (Rev. 3), Design, Inspection and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants.

1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

B 3.7 PLANT SYSTEMS

B 3.7.15 Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the fuel storage 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 fuel storage 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 15.7.4 (Ref. 3).

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel in the fuel building, the water level in the fuel storage pool is an initial condition design parameter in the analysis of the fuel handling accident as postulated by ~~Reg. Guide 1.25~~ Regulatory Guide 1.183 (Ref. 4). Irradiated fuel being moved is assumed to be from a reactor core which has been subcritical for at least 72 hours. A minimum water level of 23 feet ~~(Regulatory Position C.1.c of Ref. 4)~~ allows a decontamination factor of ~~100 (Regulatory Position C.1.g) of Ref. 4~~ 200 to be used in the accident analysis for iodine. This relates to the assumption that ~~99%~~ 99.5% of the totaliodine releas the pellet to cladding gap of the damaged rods is retained by the fuel storage pool water. The fission product release point is assumed to be at the point of impact at the top of the spent fuel storage racks. ~~The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 4).~~

The fuel handling accident inside the fuel building is described in Reference 3. With a minimum water level of 23 feet and a minimum decay time of 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 and control room doses are maintained well within the limits of 10 CFR ~~100~~ 50.67 and Regulatory Guide 1.183 (Refs. ~~4 and 5~~ 5 and 4).

The fuel storage pool water level satisfies Criterion 2 and 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (Continued)

REFERENCES

1. FSAR, Section 9.1.2, Spent Fuel Storage.

2. FSAR, Section 9.1.3, Fuel Pool Cooling and Cleanup System.

3. FSAR, Section 15.7.4, Fuel Handling Accidents.

4. Regulatory Guide ~~1.25, Rev. 0, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility For Boiling and PressurizedWaterReactors~~.

5. 10 CFR ~~100.11~~.

~~6.~~ ~~NUREG-0800, Section 15.7.4, Standard Review Planfor the Review of Safety Analysis Reports for Nuclear Power Plants~~.

50.67

1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000

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 productsin 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.

inventory

This limit is lower than the activity value that might be expected from a 1 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 SpecificActivity"). 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 thereactor cool 2-hour GE. ~~Most of the iodine isotopes have short half lives, (i.e., < 20 hours).~~

Operating a unit at the allowable secondary coolant specific activity will assure that the potential ~~2 hour~~ exclusion area boundary (EAB) exposure is ~~limited to a small fraction of~~ the 10 CFR ~~100~~ (Ref. 1) limits.

within 50.67

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the FSAR, Chapter 15.1.5 (Ref. 2) assu m and Regulatory Guide 1.183 (Ref. 3) coolant specific activity to have aradioactiv greater than 0.10 μCi/gm DOSE EQUIVAL s 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 ~~a small fraction of~~ the unit EAB limits (Ref. ~~1) for wholebody and thyroid dose rates~~.

With the loss of offsite power, the remaining steam generat 3) for TEDE. available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric steam dump valves (ASDs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam ~~generator is~~ generators are assumed to discharge steam and any entrained activity through the MSSVs and ~~(ASDs)~~ ASDs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be $\leq$ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to ~~a small fraction of~~ within the required limit (Ref. ~~1~~ 3).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, both the RCS and the steam generators are at reduced pressure or are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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 15.1.5, Steam System Piping Failure.

3. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABIL and control room ~~sure~~ means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the 10 CFR 50 Appendix J leakage criteria and tests are not required.

50.67 and Regulatory Guide 1.183

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~~. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

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. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment and if closed, the containment equipment hatch must be held in place by at least four bolts. Alternatively, the equipment hatch can be open provided it can be installed with a minimum of four bolts holding it in place. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." The personnel air lock is nominally a right circular cylinder, approximately 10 ft in diameter with a door at each end. The emergency air lock is approximately 5 ft 9 in inside diameter with a 2 ft 6 in door at each end. The doors are normally interlocked toprevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both

(continued)

BASES

BACKGROUND (continued)

"Direct access from the containment atmosphere" is defined as: Theaction of the containment atmosphere proceeding from containment to the outside atmosphere without deviation or interruption and having no impairing element.

50.67

limits

APPLICABLE SAFETY ANALYSES

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). The fuel handling accident (in containment) analyzed in Reference 2 consists of dropping a single irradiated fuel assembly onto other irradiated fuel assemblies. The requirements of LCO 3.9.7, "Refueling Pool Water Level," and the minimum decay time of 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.

Per Regulatory Guide 1.183 (Ref. 3), the

are approximately

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

50.67

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge penetrations and the personnel air lock, the emergency air lock, and the equipment hatch, which must be capable of being closed. For the OPERABLE containment purge penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge Isolation System to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit. During CORE ALTERATIONS or during movement of irradiated fuel assemblies within containment, Containment Purge Isolation valves are OPERABLE if they are capable of being closed by manual actuation. For the containment personnel air lock and emergency air lock, one air lock door must be capable of being closed. Thus both containment personnel air lock and emergency air lock doors may be open during movement of irradiated fuel assemblies within containment or CORE ALTERATIONS, provided an air lock door for each air lock is capable of being closed. Administrative controls ensure that 1) appropriate personnel are aware

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.4.3

This Surveillance demonstrates that each containment purge isolation valve actuates to its isolation position on manual initiation. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the INSERVICE TESTING PROGRAM requirements. These Surveillances will ensure that the valves are capable of being manually closed after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

REFERENCES

1. Amendment 114 to Facility Operating License No. NPF-30, Callaway Unit 1, dated July 15, 1996.

2. FSAR, Section 15.7.4.

3. ~~NUREG-0800, Section 15.7.4, Rev. 1, July 1981~~.

4. Amendment 138 to Facility Operating License No. NPF-30, Callaway Unit 1, dated September 26, 2000.

Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Pool Water Level

approximately

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 fuel transfer canal, refueling pool 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. 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 and acceptance in~~ Reference ~~6~~.

1.183 1 200 50.67

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling pool is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide ~~1.25~~ (Ref. 1). The reactor is assumed to have been subcritical for 72 hours prior to movement of irradiated fuel in the reactor vessel. A minimum water level of 23 ft ~~(Regulatory Position C.1.c of Ref. 1)~~ allows a decontamination factor of ~~100~~ (Regulatory Position ~~C.1.g~~ of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that ~~99~~% of the total iodine released from the pellet to cladding gap of the damaged rods is retained by the refueling pool water. In addition, analyses for the accident in the reactor building, the dropped assembly is assumed to damage 20% of the rods of a different assembly. The fission product release point is assumed to be at the point of impact at the top of the reactor vessel flange. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

99.5 B.2

and control room

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 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 the limits of 10 CFR ~~100~~ (Refs. ~~4, 5, and 6~~).

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

50.67 and Regulatory Guide 1.183 3 and 1

(continued)

BASES (Continued)

1

LCO	A minimum refueling pool water level of 23 ft above the top of the reactor vessel flange is required to ensure that the radiological consequencesof a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference ~~3~~.
APPLICABILITY	LCO 3.9.7 is applicable when moving irradiated fuel assemblies within containment. Proper removal and reinstallation of the upper internalswith irradiated fuel in the vessel does not constitute movement of irradiated fuel, therefore, this LCO is not applicable during installation and removal of the reactor vessel upperinternals. 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	<u>A.1</u> With a water level of < 23 ft above the top of the reactor vessel flange, 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	<u>SR 3.9.7.1</u> 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.

(continued)

BASES (Continued)

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.

2. FSAR, Section 15.7.4.

3. NUREG-0800, Rev. 1, July 1981, Section 15.7.4.

~~4.~~ 10 CFR 100.11.

~~5.~~ Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-828, Radiological Consequences of a Fuel Handling Accident, December 1971.

~~6.~~ NUREG-0830, Safety Evaluation Report, Callaway Plant, Unit No. 1, October 1981, Section 15.4.6.

10 CFR 50.67, "Accident Source Term."

Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.