B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for any LOCA that reduces RCS pressure to below the accumulator pressure.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection (SI) water.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor operated isolation valve and two check valves in series.

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BASES

BACKGROUND (continued)

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 1). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

The limiting large break LOCA is a double ended guillotine break at the discharge of the reactor coolant pump. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time,

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APPLICABLE SAFETY ANALYSES (continued)

the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and high head safety injection (HHSI) pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the HHSI pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 2) will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged.

Accumulator tank size and accumulator water volume directly affect the volume of nitrogen cover gas whose expansion produces the passive injection and thus affects injection rate. The amount of water is also important since the accumulator water which has not been injected and bypassed during blowdown is primarily responsible for filling the lower plenum (refill) and downcomer. The elevation head of the downcomer water provides the driving force for core reflooding (Ref. 3).

For large break LOCAs, changes in accumulator water volume can result in either improved or worsened analysis results; therefore, a nominal accumulator water volume of 795 cubic feet is modeled in the analysis (Ref. 3).

For small break LOCAs, changes in accumulator water volume are not significant because the clad temperature transient is terminated before the accumulators empty; therefore, a nominal accumulator water volume of 795 cubic feet is modeled in the analysis (Ref. 3).

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The large and small break LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure limit prevents injection of nitrogen into the RCS, accumulator relief valve actuation, and ultimately preserves accumulator integrity.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 3 and 4).

The accumulators satisfy Criterion 3 of 10 CFR 50.36.

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 2000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the

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BASES

APPLICABILITY (continued)

ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated discharge isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

Note 1 provides an exception to SR 3.5.1.1 and SR 3.5.1.5 and specifies that all accumulator discharge isolation valves may be closed and energized for up to 8 hours during the performance of reactor coolant system hydrostatic testing. This allowance is necessary because limits imposed by the Pressure/Temperature Limits for a hydrostatic leak test, could, in some instances, require reactor coolant system hydrostatic testing above 350°F (Mode 3). This allowance is acceptable because hydrostatic testing is performed in MODE 3 when the need for the accumulators is reduced and Note 1 limits the duration to the time needed to perform required testing.

Note 2 also provides an exception to SR 3.5.1.1 and SR 3.5.1.5 and specifies that one accumulator discharge isolation valve may be closed and energized in MODE 3 for up to 8 hours for accumulator check valve leakage testing. This allowance is acceptable because testing is limited to MODE 3 when the need for the accumulators is reduced and Note 2 limits the duration to the time needed to perform required testing.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of

(continued)

BASES

ACTIONS

A.1 (continued)

the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators do not discharge following a large main steam line break. Even if they do discharge, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated 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 MODE 3 within 6 hours and reactor coolant pressure reduced to $\leq$ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

D.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Each accumulator valve should be verified to be fully open every 12 hours. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If a discharge isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Every 12 hours, borated water volume and nitrogen cover pressure are verified for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or

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SURVEILLANCE REQUIREMENTS

SR 3.5.1.4 (continued)

inleakage. Sampling the affected accumulator within 6 hours after an increase of 8.4 cubic feet will identify whether inleakage has caused a reduction in boron concentration to below the required limit. Considering the nominal accumulator volume of 795 cubic feet of water, inleakage of 8.4 cubic feet of pure water would result in a boron concentration reduction of approximately 1%. An increase in the accumulator volume of 8.4 cubic feet causes a change of approximately 10% in the indicated accumulator level. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), because the water contained in the RWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 4).

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator discharge isolation valve operator when the reactor coolant system pressure is ≥ 2000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated discharge isolation valves when reactor coolant system pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns. Should closure of a valve occur, the SI signal provided to the valves would open a closed valve in the event of a LOCA.

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BASES (continued)

REFERENCES

1. FSAR, Chapter 6.
2. 10 CFR 50.46.
3. FSAR, Chapter 14.
4. NUREG-1366, February 1990.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS – Operating

BASES

BACKGROUND

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

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the recirculation and containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the recirculation sump or containment sump for cold leg recirculation. After between 14.3 and 24 hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

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BASES

BACKGROUND (continued)

The ECCS FUNCTION is provided by three separate ECCS systems: high head safety injection (HHSI), residual heat removal (RHR) injection, and containment recirculation. Each ECCS system is divided into subsystems as follows:

- HHSI System is divided into three 50% capacity subsystems (i.e., HHSI 31, 32 and 33) which share two pump discharge headers (i.e., 31 and 33). Each HHSI subsystem consists of one pump as well as associated piping and valves to transfer water from the suction source to the core. HHSI subsystem 32 is aligned to inject using the flow path associated with both HHSI subsystem 31 and 33. If either HHSI pump 31 or 33 fails to start or achieve required discharge pressure, HHSI pump 32 will inject via the header associated with the failed pump. If all three HHSI pumps start, flow from HHSI pump 32 will be divided between header 31 and 33. Note that the HHSI pumps have a shutoff head of approximately 1500 psig. Therefore, IP3 is classified as a low head safety injection plant.

- RHR injection System is divided into two 100% capacity subsystems. Each ECCS RHR subsystem consists of one RHR pump and one RHR heat exchanger as well as associated piping and valves to transfer water from the suction source to the core. Although either RHR heat exchanger may be credited for either RHR subsystem, one RHR heat exchanger must be OPERABLE for each OPERABLE RHR injection subsystem.

- Containment Recirculation is divided into two 100% capacity subsystems. Each subsystem consists of one Containment Recirculation pump and one RHR heat exchanger as well as associated piping and valves to transfer water from the suction source to the core. Although either RHR heat exchanger may be credited for either Recirculation subsystem, one RHR heat exchanger must be OPERABLE for each OPERABLE Containment Recirculation subsystem.

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BASES

BACKGROUND (continued)

- The three ECCS systems (3 HHSI, 2 RHR and 2 Recirculation) are grouped into three trains (5A, 2A/3A and 6A) such that any 2 of the 3 trains are capable of meeting all ECCS capability assumed in the accident analysis. Each ECCS train consists of the following:

  a. ECCS Train 5A includes subsystems HHSI 31 and containment recirculation 31;

  b. ECCS Train 2A/3A includes subsystems HHSI 32 and RHR 31; and,

  c. ECCS Train 6A includes subsystems HHSI 33, RHR 32, and containment recirculation 32.

The ECCS trains use the same designation as the Safeguards Power Trains required by LCO 3.8.9, Distribution Systems - Operating, with Safeguards Power Train 5A supported by DG 33, Safeguards Power Train 2A/23 supported by DG 31, Safeguards Power Train 6A supported by DG 32.

The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the high head safety injection pumps, the RHR pumps, heat exchangers, and the containment recirculation pumps. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from different trains to achieve the required 100% flow to the core.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the HHSI and RHR pumps. The discharge from the HHSI and RHR pumps feed injection lines to each of the RCS cold legs. Control valves are set to balance the

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BASES

BACKGROUND (continued)

HHSI flow to the RCS. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

During the recirculation phase of LOCA recovery, the containment recirculation pumps take suction from the containment recirculation sump and direct flow through the RHR heat exchangers to the cold legs. The RHR pumps can be used to provide a backup method of recirculation in which case the RHR pump suction is transferred to the containment sump. The RHR pumps then supply recirculation flow directly or supply the suction of the HHSI pumps. Initially, recirculation is through the same paths as the injection phase. Subsequently, recirculation injection is split between the hot and cold legs.

The ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of HHSI pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

The ECCS subsystems, except for the containment recirculation subsystems, are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

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BASES

BACKGROUND (continued)

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The HHSI pumps are credited in a small break LOCA event. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite power and a single failure disabling one EDG; and

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BASES

APPLICABLE SAFETY ANALYSES (continued)

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one EDG.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the HHSI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36.

LCO

In MODES 1, 2, and 3, three ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting any one train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, the ECCS consists of the following:

a. ECCS Train 5A includes HHSI subsystem 31 and containment recirculation subsystem 31;

b. ECCS Train 2A/3A includes HHSI subsystem 32 and RHR subsystem 31; and,

(continued)

BASES

LCO (continued)

c. ECCS Train 6A includes HHSI subsystem 33, RHR subsystem 32, and containment recirculation subsystem 32.

Each HHSI subsystem consists of one pump as well as associated instrumentation, piping and valves to transfer water from the suction source to the core. HHSI subsystem 32 is OPERABLE when capable of injecting using the flow paths associated with HHSI subsystem 31 and 33. Each ECCS RHR subsystem consists of one RHR pump and one RHR heat exchanger as well as associated instrumentation, piping and valves to transfer water from the suction source to the core. Although either RHR heat exchanger may be credited for either RHR subsystem, one RHR heat exchanger must be OPERABLE for each OPERABLE RHR injection subsystem.

Each containment recirculation subsystem consists of one Containment Recirculation pump and one RHR heat exchanger as well as associated instrumentation piping and valves to transfer water from the suction source to the core. Although either RHR heat exchanger may be credited for either Recirculation subsystem, one RHR heat exchanger must be OPERABLE for each OPERABLE Containment Recirculation subsystem. Note that Recirculation pump OPERABILITY requires the functional availability of the associated auxiliary component cooling water pump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the HHSI and RHR pumps and their supply header to each of the four cold leg injection nozzles (8 cold leg injection nozzles for the HHSI pumps). In the long term, this flow path may be switched to take its supply from the containment recirculation sump using the containment recirculation pumps or, alternately, the containment sump using the RHR pumps to supply its flow to the RCS hot and cold legs, either directly into the RCS or via the HHSI pumps.

The flow path for each train must maintain its designed independence to ensure that no single failure can disable more than one ECCS train (except as described in Reference 5).

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BASES

LCO (continued)

As indicated in Note 1, the SI flow paths may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. This is acceptable because the flow paths are readily restorable from the control room or the valves are opened under administrative controls that ensure prompt closure when required. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room.

As indicated in Note 2, operation in MODE 3 with ECCS trains made incapable of injecting pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is necessary for plants with an LTOP arming temperature at or near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be made incapable of injecting at and below the LTOP arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to restore the inoperable pumps to OPERABLE status.

APPLICABILITY

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

This LCO is only applicable in MODE 3 and above. Below MODE 3, system functional requirements are relaxed as described in LCO 3.5.3, "ECCS – Shutdown."

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant

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BASES

APPLICABILITY (continued)

Circulation – High Water Level," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level."

ACTIONS

A.1

With one or more trains inoperable and any two HHSI pumps, any one RHR pump, and any one Containment Recirculation pump OPERABLE (i.e., 100% of the ECCS capability assumed in the accident analysis is available), the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 4) and is a reasonable time for repair of many ECCS components. If 100% of the ECCS capability assumed in the accident analysis is not OPERABLE, entry into LCO 3.0.3 is required.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one pump in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different pumps, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that 100% of the ECCS flow equivalent to two OPERABLE ECCS trains remains available. This allows increased flexibility in plant operations under circumstances when pumps in redundant trains are inoperable.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 4) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

(continued)

BASES

ACTIONS

A.1 (continued)

Reference 5 describes situations in which one component, such as the valves governed by SR 3.5.2.1, can disable more than one ECCS train. With one or more component(s) inoperable such that 100% of the flow equivalent for HHSI, RHR and Containment Recirculation is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status within the associated 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 MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render more than one ECCS train inoperable. Securing these valves in position by removal of power or by key locking the control in the correct position ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 5, that can disable the function of more than one ECCS train and invalidate the accident analyses. A 12 hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.4 and SR 3.5.2.5

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. Note that the Containment Recirculation system is a manually initiated system and is not included as part of this SR. Additionally, this Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.6

Realignment of valves in the flow path on an SI signal is necessary for proper ECCS performance. These valves have stops to allow proper positioning for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. Therefore, an improperly positioned valve could result in the inoperability of more than one injection flow path. The stops are set based on the results of the most recent ECCS operational flow test. The 24 month Frequency is based on the reasons stated in SR 3.5.2.4 and SR 3.5.2.5.

SR 3.5.2.7

Periodic inspections of each containment and recirculation sump suction inlet ensure that each is unrestricted and stays in

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.7 (continued)

proper operating condition. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency is sufficient to detect abnormal degradation and is confirmed by industry operating experience.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. FSAR, Section 14.
4. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
5. IE Information Notice No. 87-01.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS – Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2,"ECCS – Operating," is applicable to these Bases, with the following modifications.

In MODE 4, one ECCS residual heat removal (RHR) subsystem and one ECCS Recirculation subsystem are required.

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) or the containment or recirculation sump can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Only one ECCS residual heat removal (RHR) subsystem and one ECCS Recirculation subsystem are required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of 10 CFR 50.36.

LCO

In MODE 4, one ECCS residual heat removal (RHR) subsystem and one ECCS Recirculation subsystem are required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

(continued)

BASES (continued)

LCO (continued)

In MODE 4, ECCS requirements may be met using containment Recirculation subsystem 31 or 32 and RHR subsystem 31 or 32.

An ECCS RHR subsystem consists of one RHR pump and one RHR heat exchanger as well as associated piping and valves and instrumentation and controls needed to transfer water from the RWST or containment sump to the core. Either RHR heat exchanger may be used with either RHR pump to meet requirements for an RHR subsystem.

A containment Recirculation subsystem consists of one Containment Recirculation pump and one RHR heat exchanger as well as associated piping, valves, instrumentation and controls needed to transfer water from the recirculation sump to the core. Note that Recirculation pump OPERABILITY requires the functional availability of the associated auxiliary component cooling water pump. Either RHR heat exchanger may be used with either recirculation pump to meet requirements for a recirculation subsystem. The same RHR heat exchanger may be used to meet requirements for both the RHR subsystem and the Recirculation subsystem.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the RHR pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, the recirculation flow path using the Recirculation sump or containment sump may be used to deliver its flow to the RCS cold legs.

This LCO is modified by a Note that allows an RHR subsystem to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4. Similarly, this Note allows an RHR subsystem to be considered OPERABLE during alignment and operation for valve testing if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows testing of certain valves in MODE 4.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2. In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS residual heat removal (RHR) subsystem and one OPERABLE ECCS Recirculation subsystem is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level."

ACTIONS

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

(continued)

BASES

ACTIONS (continued)

B.1

With no containment Recirculation subsystem OPERABLE, due to the inoperability of the pump or flow path from the recirculation sump, the plant is not prepared to provide long term cooling response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one ECCS Recirculation subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where a recirculation subsystem is not required.

C.1

When the Required Actions of Condition B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

Note: Condition C should not be entered if Condition A is applicable. Required Action C.1 does not mandate a cooldown to MODE 5 when a required ECCS RHR subsystem is not OPERABLE (i.e., Condition A) because plant cooldown may not be possible with inoperable RHR subsystems.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

The applicable references from Bases 3.5.2 apply.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling cavity during refueling, to the ECCS to fill accumulators, and to the ECCS and the Containment Spray System during accident conditions.

The RWST supplies the ECCS and the Containment Spray System through separate supply headers during the injection phase of a loss of coolant accident (LOCA). Motor operated isolation valves are provided to isolate the RWST from the ECCS subsystems once the system has been transferred to the recirculation mode. The switchover to the cold leg recirculation phase is manually initiated when the RWST level has reached the low-alarm setpoint and sufficient coolant inventory to support pump operation in recirculation mode is verified to be in the containment. Use of a single RWST to supply all of the injection trains of the ECCS and Containment Spray System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events.

During normal operation in MODES 1, 2, and 3, the high head safety injection (HHSI) and residual heat removal (RHR) pumps are aligned to take suction from the RWST.

The ECCS and Containment Spray System pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions.

This LCO ensures that:

a. The RWST contains sufficient borated water to support the ECCS during the injection phase;

(continued)

BASES

BACKGROUND (continued)

b. Sufficient water volume exists in the recirculation sump or the containment sump to support continued operation of the ECCS and Containment Spray System pumps at the time of transfer to the recirculation mode of cooling; and

c. The reactor remains subcritical following a LOCA or MSLB.

Insufficient water in the RWST could result in insufficient cooling capacity when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment due to improper pH in the sumps.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS - Operating"; B 3.5.3, "ECCS - Shutdown"; and B 3.6.6, "Containment Spray System and Containment Fan Cooler System." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the accident analyses.

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered.

For a large break LOCA analysis, the minimum water volume limit of 195,800 gallons and the lower boron concentration limit of 2400 ppm are used to compute the post LOCA sump boron

(continued)

APPLICABLE SAFETY ANALYSES (continued)

concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The RWST level required by Technical Specifications includes allowances for instrument accuracy, the unusable volume in the RWST, and the maximum volume expected to remain in the RWST when the plant is switched from the injection to recirculation modes of operation.

The upper limit on boron concentration of 2600 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 35°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The upper temperature limit of 110°F is used in the LOCA containment integrity analysis. Exceeding this temperature will result in higher containment pressures due to reduced containment spray cooling capacity. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

Following a LOCA, switchover from the injection phase to the recircuation phase must occur before the RWST empties to prevent damage to the pumps and a loss of cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment to support recirculation pump suction. Furthermore, early switchover must not occur to ensure that sufficient borated water is injected from the RWST.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The IP3 ESFAS design does not include automatic switchover from the safety injection mode to the recirculation mode of operation based on low level in the RWST coincident with a safety injection signal. This function is performed manually by the operator who must be alerted by redundant RWST low level alarms. The switchover to the cold leg recirculation phase is manually initiated when the RWST level has reached the low alarm setpoint and sufficient coolant inventory to support pump operation in recirculation mode is verified to be in the containment.

The RWST low level alarm setpoint has both upper and lower limits. The upper limit is set to ensure that switchover does not occur until there is adequate water inventory in the containment to provide ECCS pump suction. (This is confirmed by recirculation and/or containment sump level indication.) The lower limit is set to ensure switchover occurs before the RWST empties, to prevent ECCS pump damage.

Requiring 2 channels of RWST low level alarm ensures that the alarm function will be available assuming a single failure of one channel.

The RWST satisfies Criterion 3 of 10 CFR 50.36.

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the recirculation sump and the containment sump to support ECCS pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water level, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and Containment Spray System OPERABILITY

(continued)

BASES

APPLICABILITY (continued)

requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level."

ACTIONS

A.1

With RWST boron concentration or borated water temperature not within limits of SR 3.5.4.3 and SR 3.5.4.1, respectively, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

B.1

Condition B applies when one channel of RWST low level alarm is inoperable. Required Action B.1 requires restoring the inoperable channel to OPERABLE status within 7 days. The 7 day Completion Time for restoration of redundancy to the alarm function is needed because the IP3 ESFAS design does not include automatic switchover from the safety injection mode to the recirculation mode of operation based on low level in the RWST coincident with a safety injection signal. This function is performed manually by the operator who is alerted by the RWST low level alarm as the primary indicator for determining the time for the switchover. The 7 day Completion Time for restoration of redundancy for this alarm function is acceptable because of the remaining alarm channel and the availability of containment and recirculation sump level indication in the containment.

(continued)

BASES

ACTIONS (continued)

C.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

In this Condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

D.1 and D.2

If the RWST cannot be returned to OPERABLE status within the associated 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 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be within the limits assumed in the accident analyses band. This Frequency is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating limits of the RWST. With ambient air temperatures within the band, the RWST temperature should not exceed the limits.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.2

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS System pump operation on recirculation.

Since the RWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.3

The boron concentration of the RWST should be verified every 31 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST level is normally stable, a 31 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.4

Performance of the CHANNEL CHECK every 7 days ensures that a gross failure of the RWST level instruments has not occurred. A CHANNEL CHECK is normally the comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same channel should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure: thus, it is key to verifying that the RWST level instruments continue to operate properly between each CHANNEL CALIBRATION.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.4.4 (continued)

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the RWST level instrument channel has drifted outside the limit. If the channels are within criteria, it is an indication that the RWST level instrument channels are OPERABLE.

The frequency of 7 days is based on operating experience that demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of displays associated with the LCO required RWST level instruments.

SR 3.5.4.5

A CHANNEL CALIBRATION of the RWST level indicating switch is performed at least every 184 days. CHANNEL CALIBRATION is a complete check of the level indicating switch loop including the required alarm. The test verifies the RWST level indicating switch responds to RWST level within the required range and accuracy. The test also verifies that the RWST level indicating switch will cause the low level alarm to annunciate at ≥10.5 feet and ≤12.5 feet to ensure the operator is alerted to start the switchover to the recirculation mode during accident conditions. The frequency is based on operating experience and previous license commitments.

SR 3.5.4.6

A CHANNEL CALIBRATION of the RWST level transmitter is performed at least every 18 months. CHANNEL CALIBRATION is a complete check of the RWST level transmitter loop including the required alarm. The test verifies the RWST level transmitter responds to RWST level within the required range and accuracy.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.4.6 (continued)

The test also verifies that the RWST level transmitter will cause the low level alarm to annunciate at ≥10.5 feet and ≤12.5 feet to ensure the operator is alerted to start the switchover to the recirculation mode during accident conditions. The frequency is based on operating experience and previous license commitments.

REFERENCES

1. FSAR, Chapter 6 and Chapter 14.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the concrete reactor building, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA), in particular, a Main Steam Line Break (MSLB) inside containment or a Loss of Coolant Accident (LOCA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

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

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

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

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

(continued)

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. The equipment hatch is properly closed; and

d. The Isolation Valve Seal Water (IVSW) system is OPERABLE, except as provided in LCO 3.6.9.

e. The Weld Channel and Penetration Pressurization System is OPERABLE, except as provided in LCO 3.6.10.

APPLICABLE SAFETY ANALYSES

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

The DBAs that result in a challenge to containment 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 0.1% of containment air weight per day assuming the proper functioning of the Isolation Valve Seal Water System but without benefit of the Weld Channel and Penetration Pressurization System (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 DBAs (LBLOCA or MSLB). The

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 0.1% of containment air weight per day in the safety analysis at P_a which is specified in Specification 5.5.15, Containment Leakage Rate Testing Program.

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

The containment satisfies Criterion 3 of 10 CFR 50.36.

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required leakage test in accordance with requirements in Specification 5.5.15, Containment Leakage Rate Testing Program. At this time, the applicable leakage limits specified in the Containment Leakage Rate Testing Program must be met.

Compliance with this LCO will ensure a containment configuration, including the equipment hatch, that is structurally sound and that will limit leakage to less than the leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air locks (LCO 3.6.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of $1.0\ L_a$.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES.

(continued)

BASES

APPLICABILITY (continued)

Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment 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 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock leakage limits specified in LCO 3.6.2 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

startup after performing the Containment Leakage Rate Testing Program leakage test is required to be $\leq 0.6\ L_a$ for combined Type B and C leakage and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. FSAR, Chapter 14.
3. FSAR, Chapter 6.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

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

Each air lock is a cylinder with a door at each end. One of the two air locks is designed as a part of the containment structure and the other is designed as an integral part of the containment equipment hatch but otherwise the two air locks function identically. 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 air lock door and the equipment hatch is designed with double gasketed seals to permit pressurization between the gaskets. The double gasketed seals are normally continuously pressurized above accident pressure. Finally, 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) and local leakage rate testing capability is available to ensure containment integrity is being maintained.

The doors are interlocked to prevent simultaneous opening of the inner and outer door. This interlock is a requirement for OPERABILITY. 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 personnel air lock is provided with limit switches on both doors that provide control room indication when an airlock door is not fully closed.

(continued)

BASES

BACKGROUND (continued)

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 unit safety analyses.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident. In the analysis of each of these accidents, 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 0.1% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a = 0.1% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = 42.40 psig following a DBA (LBLOCA or MSLB). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36.

LCO

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

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach

(continued)

BASES

LCO (continued)

of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

The program established by Specification 5.15, "Containment Leakage Rate Test Program," which conforms to NEI 94-01, Section 10.2.2 (Ref. 3) for Containment Air Locks, requires that air lock doors opened during periods when containment integrity is required must be tested within 7 days after being opened. For Indian Point 3, which has air locks with testable seals, this requirement is satisfied in accordance with ANSI/ANS-56.8-1994 "Containment System Leakage Testing Requirements," (Ref. 4) by testing the seals (i.e., verifying that seals re-pressurize to the required pressure after an airlock door is closed). Pressurization of air lock seals is not required for air lock OPERABILITY except as needed to satisfy testing requirements after being opened.

APPLICABILITY

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

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. When the inner door is inoperable, it is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means

(continued)

BASES

ACTIONS (continued)

there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

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

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

(continued)

BASES

ACTIONS

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

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required on a periodic basis to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment is entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

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BASES

ACTIONS (continued)

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

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable, since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

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BASES

ACTIONS

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

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time unless Condition C is exited in accordance with LCO 3.0.2 (i.e., one door is made OPERABLE). The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the inoperable containment air lock 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 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 1), required by Specification 5.5.15, Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1 (continued)

initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by Specification 5.5.15, Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria that is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. The 24 month Frequency is based on the need to perform this Surveillance under conditions that apply during a plant

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2 (continued)

outage, and the potential for loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. The 24 month Frequency for the interlock is justified based on generic operating experience. The Frequency is based on engineering judgment and is considered adequate given that the interlock is not normally challenged during the use of the airlock.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.

2. FSAR, Section 6.6.

3. NEI 94-01, Section 10.2.2.

4. ANSI/ANS-56.8-1994, "Containment System Leakage Testing Requirements."

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a safety injection signal. The Phase "A" isolation signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Phase "B" isolation occurs upon receipt of a containment pressure High-High signal and isolates the remaining process lines, except systems required for accident mitigation. In addition to the isolation signals listed above, the Containment purge supply and exhaust isolation valves (FCV-1170, FCV-1171, FCV-1172, and FCV-1173) and the containment pressure relief isolation valves (PCV-1190, PCV-1191, and PCV-1192) close when high radiation levels are detected by the Containment Air Particulate Monitor (R-11) or Containment Radioactive Gas Monitor (R-12). Containment purge and containment pressure relief are also isolated when high radiation levels are detected in the plant vent. As a result, the

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BASES

BACKGROUND (continued)

containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analyses. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analyses will be maintained.

Containment Purge System (36 inch purge valves)

The Containment Purge System, consisting of purge supply and exhaust isolation valves FCV-1170, FCV-1171, FCV-1172, and FCV-1173, operates to supply outside air into the containment for ventilation and cooling or heating and may also be used to reduce the concentration of noble gases within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. Because of their large size, the 36 inch purge valves are not qualified for automatic closure from their open position under DBA conditions. Therefore, the 36 inch purge valves must be maintained sealed closed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

Containment Pressure Relief Line (10 inch valves)

The Containment Pressure Relief Line consisting of pressure relief isolation valves PCV-1190, PCV-1191, and PCV-1192, operates to:

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

b. Equalize internal and external pressures.

Since the valves used in the Containment Pressure Relief Line are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3, and 4. Containment pressure relief line

(continued)

BASES

BACKGROUND (continued)

isolation valve opening is limited by mechanical stops so that opening angle is limited to an angle at which analysis indicates the valve will operate against containment accident pressures. Additionally, pressure relief isolation valve opening must be limited to the time necessary for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.

The containment pressure relief line is isolated during CORE ALTERATIONS and movement of irradiated fuel inside containment in accordance with requirements established in LCO 3.9.3, Containment Penetrations.

APPLICABLE SAFETY ANALYSES

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

The DBA that results in a release of radioactive material within containment is a loss of coolant accident (LOCA) (Ref. 1). In the analyses for this accident, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves are minimized. The safety analyses assume that the 36 inch purge valves are sealed closed at event initiation.

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

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The containment purge supply and exhaust isolation valves (FCV-1170, FCV-1171, FCV-1172, and FCV-1173) may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. In this case, the single failure criterion remains applicable to the containment purge valves due to failure in the control circuit associated with each valve. Again, the purge system valve design precludes a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.

Sealed closed barriers include blind flanges and sealed closed isolation valves including closed manual valves, closed remote-manual valves, and closed automatic valves which remain closed after a loss-of-coolant accident. Sealed closed barriers may be used in place of any automatic isolation valve. The term sealed closed, as applied to containment isolation valves, is not intended to describe leak tightness. Sealed closed isolation valves must be under administrative controls that assure the valve cannot be inadvertently opened. Administrative controls includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator (Ref. 3).

The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36.

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 36 inch purge valves must be maintained sealed closed.

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BASES

LCO (continued)

The valves covered by this LCO are listed in the FSAR (Ref. 2). The passive isolation devices are shown on drawings in the FSAR. The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact (Ref. 3).

Manually operated containment isolation valves on essential lines that are required to be open, at least for a time, during post accident conditions are OPERABLE if they can be closed in accordance with design assumptions. Essential lines are those lines required to mitigate an accident, or which, if unavailable, could increase the magnitude of the event. Also, those lines which, if available, would be used in the short term (24 to 36 hours) to restore the plant to normal operation following an event which has resulted in containment isolation (Ref. 4).

This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

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

ACTIONS

The ACTIONS are modified by Note 1 which allows penetration flow paths, except for 36 inch purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment

(continued)

BASES

ACTIONS (continued)

isolation is indicated. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust from the containment atmosphere to the environment, the penetration flow path containing these valves may not be opened under administrative controls.

The normally stationed control room operator satisfies the requirement for a dedicated operator for any non-automatic, remotely operated CIV that is opened intermittently from the control room (Ref. 6). Additionally, a dedicated operator is not required for manually operated CIVs required to be open both during normal plant operations and during a LOCA. A dedicated operator is not required at the valve when the RHR Suction isolation valve, AC-732, is open to support operation of the RHR system for shutdown cooling (Ref. 6). Normally open, manual CIVs are used for isolation of closed systems within the containment that are missile protected and are seismic Class I at least up to and including the isolation valves.

Note 2 has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

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

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

The ACTIONS are further modified by Note 5 and Note 6, which ensures appropriate remedial actions are taken if required IVSW

(continued)

BASES

ACTIONS (continued)

or WC&PPS supply to a penetration flowpath is inoperable. Note 5 and Note 6 direct entry into the applicable Conditions and Required Actions of LCO 3.6.9 and LCO 3.6.10, as appropriate.

A.1 and A.2

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

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. This action involves verification, through a system walkdown, that isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment e.g., one of the three containment

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

pressure relief isolation valves, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two or more containment isolation valves. Although most penetrations have two containment isolation valves, the term "two or more" is used so that Condition A includes penetrations such as the pressure relief line penetration which has three valves in series. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides the appropriate actions.

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices once they have been verified to be in the proper position, is small.

B.1

With two or more containment isolation valves in one or more penetration flow paths inoperable, except for containment bypass leakage or hydrostatically tested valve leakage, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with

(continued)

BASES

ACTIONS

B.1 (continued)

Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two or more containment isolation valves. Although most penetrations have two containment isolation valves, the term "two or more" is used so that Condition B includes penetrations such as the pressure relief line penetration which has three valves in series. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier, other than the closed system, that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange (Ref. 3). A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system. The closed system must meet the requirements of Reference 3.

Required Action C.2 is modified by a Note that applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

D.1

With the containment bypass leakage rate not within limit of SR 3.6.3.9, the assumptions of the safety analyses are not met. Therefore, the leakage must be restored to within limit within 4 hours. Restoration can be accomplished by isolating the penetration(s) that caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. When a penetration is isolated the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed

(continued)

BASES

ACTIONS

D.1 (continued)

to be the lesser actual pathway leakage of the two devices. The 4 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration(s) and the relative importance of containment bypass leakage to the overall containment function.

With the hydrostatically tested valve leakage not within limit of SR 3.6.3.10, the potential exists for flooding the Containment Recirculation Pumps during long term post-accident cooling. The 72 hour Completion Time is reasonable because of the low probability of an event occurring during this period.

E.1 and E.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each 36 inch containment purge supply and exhaust isolation valve (FCV-1170, FCV-1171, FCV-1172, and FCV-1173) is required to be verified sealed closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A containment purge valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1 (continued)

In this application, the term "sealed" has no connotation of leak tightness. The Frequency is a result of an NRC initiative, Generic Issue B-24 (Ref. 5), related to containment purge valve use during plant operations.

SR 3.6.3.2

This SR ensures that the containment pressure relief line isolation valves (PCV-1190, PCV-1191, and PCV-1192) are closed as required or, if open, open for an allowable reason. If a containment pressure relief line isolation valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the containment pressure relief line isolation valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The containment pressure relief line isolation valves are capable of closing in the environment following a LOCA as long as valve opening angle is limited in accordance with SR 3.6.3.7. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.3 (continued)

containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed or otherwise secured in the closed position because these valves were verified to be in the correct position when locked, sealed or otherwise secured.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves that are open under

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.4 (continued)

administrative controls are not required to meet the SR during the time they are open. This SR does not apply to valves that are locked, sealed or otherwise secured in the closed position because these valves were verified to be in the correct position when locked sealed or otherwise secured.

This Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4, for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.5

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses as specified in the FSAR. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.6

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.6 (continued)

need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.3.7

Verifying that each containment pressure relief line isolation valve, PCV-1190, PCV-1191, and PCV-1192, is blocked to restrict valve opening to ≤ 60 degrees, is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the pressure relief line valves must close to maintain containment leakage within the values assumed in the accident analysis. The 24 month Frequency is appropriate because the blocking devices are typically not removed.

SR 3.6.3.8

This SR ensures that manually operated containment isolation valve on essential lines are capable of being opened or closed as needed to support any accident mitigation function. Essential lines are those lines required to mitigate an accident, or which, if unavailable, could increase the magnitude of the event. Also, those lines which, if available, would be used in the short term (24 to 36 hours) to restore the plant to normal operation following an event which has resulted in containment isolation (Ref. 4). The 24 month Frequency is based on engineering judgement and plant experience with manually operated valves.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.9

This SR ensures that the combined leakage rate of all containment leakage paths is less than or equal to the specified leakage rate for those paths that are not sealed by the Isolation Valve Seal Water System or sealed by the RHR system or sealed by the service water system. This provides assurance that the assumptions in the safety analysis are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves.

This testing is performed in accordance with the requirements, Frequency and acceptance criteria required by Specification 5.5.15, Containment Leakage Rate Testing Program. This program was established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by IP3 specific approved exemptions. This program conforms to guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, dated September 1995." In the event containment isolation valve leakage results in exceeding the overall containment leakage rate, entry into the applicable Conditions and Required Actions of LCO 3.6.1 is required.

SR 3.6.3.10

The Containment Leakage Rate Testing Program includes verification that inleakage rate from the containment isolation valves sealed with service water is maintained at a level that will prevent flooding the internal recirculation pumps for the full 12-month period of post accident recirculation. This inleakage test has specific acceptance criteria ($\leq$0.36 gpm per fan cooler unit when pressurized at $\geq$1.1 P_a) specified in the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.10 (continued)

Containment Leakage Rate Testing Program and the results for this inleakage test are not counted against the acceptance criteria for the Type B and C tests that are also performed as part of the SR.

REFERENCES

1. FSAR, Section 14.
2. FSAR, Section 6.
3. Standard Review Plan Section 6.2.4.
4. FSAR, Section 5.2.
5. Generic Issue B-24.
6. Safety Evaluation Report for IP3 Amendment 195.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). The containment can withstand an internal vacuum of 3 psig. The 2.0 psig vacuum specified as an operating limit avoids any difficulties with motor cooling.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients. Cycle specific analysis results indicate that the worst case peak containment pressure could result from either a loss of coolant accident or a steam line break inside containment (Ref. 1).

The initial pressure condition used in the containment analysis was +2.5 psig. This analysis concluded that the containment design pressure of 47 psig would not be exceeded for either a major loss-of-coolant accident or for a main steam line break accident. The containment analysis results are presented in Reference 1 and the current value for peak containment pressure is listed in Specification 5.5.15, "Containment Leakage Rate Testing Program."

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The containment was also designed for an external pressure load equivalent to -3.0 psig (i.e., the containment can withstand an internal vacuum of 3 psig). The -2.0 psig specified as the Limiting Condition for Operation is based on limits associated with motor cooling.

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

Containment pressure satisfies Criterion 2 of 10 CFR 50.36.

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that motor heating concerns are addressed.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

(continued)

BASES (continued)

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If containment pressure cannot be restored to within limits 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 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. FSAR, Section 14.3

2. 10 CFR 50, Appendix K.

3. FSAR Section 3.1.8, Appendix 5A.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the Containment Spray and Cooling systems during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limits violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The upper limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The lower limit is to assure that the minimum service metal temperature of the containment liner is well above the NDT + 30°F criterion for the liner material (Ref. 3).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train each of the Containment Spray System, Residual Heat Removal System, and Containment Cooling System being rendered inoperable.

The limiting DBA for the maximum peak containment air temperature may be either a LOCA or a SLB. The initial containment average air temperature is assumed in the design basis analyses. The maximum containment air temperature and the design temperature are specified in (Ref. 1). The temperature limit is used to establish the environmental qualification operating envelope for containment. The maximum peak containment air temperature was calculated to exceed the containment design temperature for only a few seconds during the transient. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 2). Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBA LOCA or SLB.

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure may be either a

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

LOCA or a SLB. The upper temperature limit is used in this analysis to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36.

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature upper limit, the resultant peak accident temperature is maintained below the containment design temperature. As a result, the ability of containment to perform its design function is ensured.

The lower limit for containment average air temperature assures that the containment liner temperature is maintained well above the NDT temperature.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limits is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature is ≤50 °F, it must be restored within limits immediately. This required action is necessary to ensure that a sufficient margin of safety is maintained so the NDT limit is not compromised. The completion time of immediately ensures that containment temperature is restored to within limits without delay.

(continued)

BASES

ACTIONS (continued)

B.1

When containment average air temperature is greater than 130 °F, it must be restored to within limits within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

C.1 and C.2

If the containment average air temperature cannot be restored to within its limits 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 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limits ensures that containment operation remains within the limits assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere.

A representative measurement of containment air temperature requires an arithmetic average of temperatures measured at no fewer than 4 locations. Environmentally and seismically qualified RTDs mounted on the crane wall above the containment fan cooler units inlet are normally used for measuring containment ambient temperature. Portable temperature sensing equipment may also be used.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1 (continued)

The 24 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

1. FSAR, Section 14.3.
2. 10 CFR 50.49.
3. FSAR, Section 5.1.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray System and Containment Fan Cooler System

BASES

BACKGROUND

The Containment Spray System and Containment Fan Cooler System provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability 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 Fan Cooler systems 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," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1).

The Containment Spray System and Containment Fan Cooler 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 Fan Cooler System provide redundant 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. Each train includes a containment spray pump, piping and valves and is independently capable of delivering one-half of the design flow needed to maintain the post-accident containment pressure below 47 psig. The spray water is injected into the containment through spray nozzles connected to four 360 degree ring headers located in the containment dome area. Each train supplies two of the four ring headers. Each train is powered from a separate safeguards power train. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation.

(continued)

BASES

BACKGROUND (continued)

After the Refueling Water Storage Tank has been exhausted, the containment recirculation pumps or the Residual Heat Removal (RHR) pumps are used to supply the Containment Spray ring headers for the long-term containment cooling and iodine removal during the containment recirculation phase. In this configuration, the RHR heat exchangers provide the necessary cooling of the recirculated containment spray.

The Containment Spray System provides a spray of cold borated water mixed with sodium hydroxide (NaOH) from the spray additive tank into the upper regions of containment to reduce the containment pressure and temperature. Additionally, these systems reduce fission products from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump or recirculation sump water by the residual heat removal heat exchangers. Both trains of the Containment Spray System are needed to provide adequate spray coverage to meet the system design requirements for containment heat removal assuming the Fan Cooler System is not available.

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

The Containment Spray System is actuated either automatically by a containment High-High pressure signal or manually. An automatic actuation starts the two containment spray pumps, opens the containment spray pump discharge valves, and begins the injection phase. A manual actuation of the Containment Spray System requires the operator to actuate two separate push buttons on the main control board to begin the same sequence. The injection phase continues until the RWST water supply is exhausted. After the Refueling Water Storage Tank has been

(continued)

BASES

BACKGROUND (continued)

exhausted, the containment recirculation pumps or the residual heat removal (RHR) pumps may be used to supply the Containment Spray ring headers for the long-term containment cooling and iodine removal during the containment recirculation phase. In this configuration, the RHR heat exchangers provide the necessary cooling of the recirculated containment spray. The Containment Spray function in the recirculation mode may be used to maintain an equilibrium temperature between the containment atmosphere and the recirculated sump water. The Containment Spray function in the recirculation mode is controlled by the operator in accordance with the emergency operating procedures.

Containment Fan Cooler System

The Containment Fan Cooler System consists of five 20% capacity Fan Cooler Units (FCUs) located inside containment. These FCUs are used for both normal and post accident cooling of the containment atmosphere. Each FCU consists of a motor, fan, cooling coils, moisture separators, HEPA filters, carbon filters, dampers, duct distribution system, instrumentation and controls. Service water is supplied to the cooling coils to perform the heat removal function.

During normal plant operation, the moisture separators, HEPA filters and activated carbon filter assembly are isolated from the main air recirculation stream. In this configuration, service water is supplied to all five FCUs and two or more FCUs fans are typically operated to limit the ambient containment air temperature during normal unit operation to less than the limit specified in LCO 3.6.5, "Containment Air Temperature." This temperature limitation ensures that the containment temperature does not exceed the initial temperature conditions assumed for the DBAs.

In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically. Additionally, the actuation signal causes the air flow (air-steam mixture) in each FCU to be split into two parts by a bypass flow control damper that fails to a pre-set position for accident operation. A minimum of 8000 cfm is directed through the FCU filtration section (moisture separators, HEPA filters, and carbon filter assembly) with the remainder of the

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BASES

BACKGROUND (continued)

air flow bypassing the filtration section. Both the filtered and unfiltered FCU flow passes through the cooling coils. The temperature of the service water is an important factor in the heat removal capability of the fan units. The accident analysis assumes 1400 gpm of service (cooling) water with a maximum river water inlet temperature of 95° F is supplied to each FCU.

Containment Cooling and Iodine Removal Function

The containment cooling and iodine removal function is provided by either of two systems:

a) the Containment Spray System consisting of two 50% capacity trains; and,

b) The Containment Fan Cooler System consisting of five 20% capacity Fan Cooler Units (FCUs).

Requirements for Containment Spray Trains may be designated by the number of the containment spray pump or the associated safeguards power train. Containment Spray Train 31 is associated with Safeguards Power Train 5A which is supported by DG 33. Containment Spray Train 32 is associated with Safeguards Power Train 6A which is supported by DG 32.

Requirements for the five fan cooler units are designated by grouping the 5 fan cooler units into three trains based on the safeguards power train needed to support Operability. This results in the following designations:

Fan Cooler Train 5A consists of FCU 31 and FCU 33;

Fan Cooler Train 2A/3A consists of FCU 32 and FCU 34; and

Fan Cooler Train 6A consists of FCU 35.

Design assumptions regarding containment air cooling and iodine removal are met by any of the following configurations:

a) Two containment spray trains; or,

(continued)

BASES

BACKGROUND (continued)

b) Three fan cooler trains (i.e., five fan cooler units); or,

c) One containment spray train and any two fan cooler trains (i.e., at least three fan cooler units).

This last configuration, one containment spray train and two fan cooler trains, is the configuration available following the loss of any safeguards power train (e.g., diesel failure).

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Fan Cooler System limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one safeguards power train, which is the worst case single active failure and results in one train of Containment Spray and one train of Fan Coolers being rendered inoperable.

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure and temperature may result from either a LOCA or SLB, depending on the cycle specific analysis (Refs. 4 and 6). Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5 for a detailed discussion.) The analyses and evaluations assume a unit specific power level of 102% and initial (pre-accident) containment conditions of 130°F and 2.5 psig and a service water inlet temperature of 95° F. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

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 activation has been analyzed. An inadvertent spray activation results in a rapid reduction of containment pressure and is associated with the sudden cooling effect in the interior of a leak tight containment. Additional documentation is provided in the Bases for LCO 3.6.4.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-High pressure setpoint to achieving full flow through the containment spray nozzles. The Containment Spray System total response time includes diesel generator (DG) startup (for loss of offsite power), loading of equipment, containment spray pump startup, and spray line filling.

Containment cooling train performance for post accident conditions is given in References 3, 4 and 6. The result of the analysis is that accident analysis assumptions regarding containment air cooling and iodine removal are met by any of the following configurations:

a) Two containment spray trains; or,

b) Three fan cooler trains (i.e., five fan cooler units); or,

c) One containment spray train and any two fan cooler trains (i.e., at least three fan cooler units).

This last configuration, one containment spray train and two fan cooler trains, is the configuration available following the loss of any safeguards power train (e.g., diesel failure).

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-High pressure setpoint to achieving full Containment Fan Cooler 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 service water pump startup times (Ref.4).

The Containment Spray System and Containment Fan Cooler System satisfy Criterion 3 of 10 CFR 50.36.

LCO

Accident analysis assumptions regarding containment air cooling and iodine removal are met by any of the following configurations:

a) Two containment spray trains; or,

b) Three fan cooler trains (i.e., five fan cooler units); or,

c) One containment spray train and any two fan cooler trains (i.e., at least three fan cooler units).

This last configuration, one containment spray train and two fan cooler trains, is the configuration available following the loss of any safeguards power train (e.g., diesel failure).

Each Containment Spray System 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.

Each FCU consists of a motor, fan, cooling coils, moisture separators, HEPA filters, carbon filters, dampers, duct distribution system, instrumentation and controls necessary to maintain an OPERABLE flow path for the containment atmosphere through both the filtration unit and cooling coils and an OPERABLE flow path for service water through the cooling coils.

(continued)

BASES (continued)

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 cooling trains.

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

ACTIONS

A.1

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 fan cooler trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal 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 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.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

With one of the required containment fan cooler trains inoperable, the inoperable required containment fan cooler train must be restored to OPERABLE status within 7 days. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Fan Cooler 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.

D.1

With two required containment fan cooler trains inoperable, one of the required containment cooling trains must be restored to OPERABLE status within 72 hours. This allowable out of service time is acceptable because the minimum required containment cooling and iodine removal function is maintained even though this configuration is a substantial degradation from the design capability, and may be a loss of redundancy for this function.

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

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

E.1

With two containment spray trains or any combination of three or more containment spray and fan cooler trains inoperable, the unit could be in a condition outside the accident analysis. Entering this Condition represents a substantial degradation of the containment heat removal and iodine removal function. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment (check valves are inside containment) and capable of potentially being mispositioned are in the correct position. Valves in containment with remote position indication may be checked using remote position indication.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.2

Operating each containment fan cooler unit for ≥ 15 minutes ensures that all fan cooler units are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 92 day Frequency was developed considering fan coolers are operated during normal plant operation, the known reliability of the fan units and controls, the two train redundancy available, and the low probability of significant degradation of the containment fan cooler units occurring between surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.6.3

Verifying that the service water flow rate to each fan cooler unit is ≥ 1400 gpm provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 3). The 92 day Frequency was developed considering the known reliability of the Cooling Water System, the redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6.4

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment High-High pressure signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The tests are performed with the isolation valves in the spray supply lines at the containment and the spray additive tank isolation valves blocked closed.

The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.6.7

This SR requires verification that each containment fan cooler unit starts and damper re-positions to the emergency mode upon receipt of an actual or simulated safety injection signal. The 24 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the 24 month Frequency.

SR 3.6.6.8

This SR verifies that the required Fan Cooler Unit testing is performed in accordance with Specification 5.5.10, Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.8 (continued)

charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.6.9

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. 10 CFR 50, Appendix A.
2. 10 CFR 50, Appendix K.
3. FSAR, Sections 6.3 and 6.4.
4. FSAR, Section 14.3.
5. ASME, Boiler and Pressure Vessel Code, Section XI.
6. WCAP - 12269, Containment Margin Improvement Analysis for IP-3 Unit 3, Rev. 1.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Spray Additive System

BASES

BACKGROUND

The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a Design Basis Accident (DBA).

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a DBA. It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Because of its stability when exposed to radiation and elevated temperature, sodium hydroxide (NaOH) is the preferred spray additive. The NaOH added to the spray also ensures an alkaline pH of the solution recirculated from the containment sump. An alkaline pH minimizes the evolution of iodine as well as the occurrence of chloride and caustic stress corrosion on mechanical systems and components.

The Spray Additive System consists of one spray additive tank that is shared by the two trains of containment spray. Each train provides a flow path from the spray tank to a containment spray pump and consists of an eductor for each containment spray pump, valves, instrumentation, and connecting piping. Each eductor draws the NaOH spray solution from the common tank using a portion of the borated water discharged by the containment spray pump as the motive flow. The eductor mixes the NaOH solution and the borated water and discharges the mixture into the spray pump suction line. The eductors are designed to ensure that the pH of the spray mixture is between 9.0 and 10.0.

The Containment Spray System actuation signal opens the valves from the spray additive tank to the spray pump suctions after a 2 minute delay. The 35% to 38% NaOH solution is drawn into the spray pump suctions. The spray additive tank capacity provides

(continued)

BASES

BACKGROUND (continued)

for the addition of NaOH solution to all of the water sprayed from the RWST into containment via the Containment Spray System. The percent solution and volume of solution sprayed into containment ensures a long term equilibrium containment sump pH of approximately 9.0. This ensures the continued iodine retention effectiveness of the sump water during the recirculation phase of spray operation and also minimizes the occurrence of chloride induced stress corrosion cracking of the stainless steel recirculation piping.

APPLICABLE SAFETY ANALYSES

The Spray Additive System, in conjunction with the Fan Cooler System, is essential to the removal 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 100% of containment is covered by the spray (Ref. 1).

The DBA response time assumed for the Spray Additive System is the same as for the Containment Spray System (plus a 2 minute delay) and is discussed in the Bases for LCO 3.6.6, "Containment Spray and Fan Cooler System."

The DBA analyses assume that one train of the Containment Spray System is inoperable and that the spray additive is added to the remaining Containment Spray System flow path.

The Spray Additive System satisfies Criterion 3 of 10 CFR 50.36.

LCO

The Spray Additive System reduces the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume and concentration of the spray additive solution must be sufficient to provide NaOH injection into the spray flow until the Containment Spray System suction path is switched from the RWST to the recirculation sump or

(continued)

BASES

LCO (continued)	containment sump, and to raise the average spray solution pH to a level conducive to iodine removal, namely, to between 7.9 and 10.0. This pH range maximizes the effectiveness of the iodine removal mechanism without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive 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 Spray Additive System is not required to be OPERABLE in MODE 5 or 6.
ACTIONS	<u>A.1</u> If the Spray Additive 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 enhancement is reduced in this condition. The Containment Spray System and Containment Fan Cooler System are available and would remove iodine from the containment atmosphere in the event of a DBA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst case DBA occurring during this period. <u>B.1 and B.2</u> If the Spray Additive System 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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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 48 hours for restoration of the Spray Additive System in MODE 3 and 36 hours to reach MODE 5. This is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Verifying the correct alignment of Spray Additive System manual, power operated, and automatic valves in the spray additive flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.7.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The 184 day Frequency was developed based on the low probability of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.7.2 (continued)

an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and alarmed in the control room, so that there is high confidence that a substantial change in level would be detected.

SR 3.6.7.3

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

SR 3.6.7.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The test is performed with the isolation valves in the spray supply lines at the containment and the spray additive tank isolation valves blocked closed.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.7.4 (continued)

Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.7.5

To ensure that the correct pH level is established in the borated water solution provided by the Containment Spray System, flow in the Spray Additive System is verified once every 5 years. This SR provides assurance that NaOH will be introduced into the flow path upon Containment Spray System initiation. This test is satisfied by a verification of spray additive system flow without pumping any NaOH solution from the spray additive tank and without draining the spray additive tank. Water may be used in lieu of NaOH for the performance of this SR which is not intended to require the transfer of NaOH. Due to the passive nature of the spray additive flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow.

REFERENCES

1. FSAR, Chapters 6 and 14.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Hydrogen Recombiners

BASES

BACKGROUND

The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen oxygen reaction.

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA) or steam line break (SLB). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammable limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent hydrogen recombiner systems are provided. Each consists of controls located in the control room, a power supply and a recombiner. Recombination is accomplished by heating a hydrogen air mixture above 1150°F. A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.1 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Features bus, and is provided with a separate power panel and control panel.

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.1 v/o following a DBA. This control would prevent a containment wide hydrogen burn, thus ensuring the pressure and temperature assumed

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

in the analyses are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA. Hydrogen may accumulate in containment following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 3 are used to maximize the amount of hydrogen calculated.

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 2.0 v/o about 5 days after the LOCA and 3.0 v/o about 10 days later if no recombiner was functioning (Ref. 3). Initiating the hydrogen recombiners when the primary containment hydrogen concentration reaches 3.0 v/o will maintain the hydrogen concentration in the primary containment below flammability limits.

The hydrogen recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

above, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.0 v/o (Ref. 3).

The hydrogen recombiners satisfy Criterion 3 of 10 CFR 50.36.

LCO

Two hydrogen recombiners must be OPERABLE. This ensures operation of at least one hydrogen recombiner in the event of a worst case single active failure.

Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

APPLICABILITY

In MODES 1 and 2, two hydrogen recombiners are required to control the hydrogen concentration within containment below its flammability limit of 4.1 v/o following a LOCA, assuming a worst case single failure.

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations in these MODES. Therefore, hydrogen recombiners are not required in these MODES.

ACTIONS

A.1

With one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within

(continued)

BASES

ACTIONS

A.1 (continued)

30 days. In this condition, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

Required Action A.1 has been modified by a Note that states the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

B.1

If the inoperable hydrogen recombiner 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. The 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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

Performance of a system functional test for each hydrogen recombiner ensures the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR verifies that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes. After reaching 700°F, the power is increased to maximum power for approximately 2 minutes and power is verified to be ≥ 60 kW.

Operating experience has shown that these components usually pass the Surveillance when performed at the 6 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.8.2

This SR ensures there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failure involves loss of power, blockage of the internal flow, missile impact, etc.

A visual inspection is sufficient to determine abnormal conditions (e.g., loose wiring or structural connections, deposits of foreign materials, etc.) that could cause such failures. The 24 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

SR 3.6.8.3

This SR requires performance of a resistance to ground test for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.8.3 (continued)

The 24 month Frequency for this Surveillance was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

REFERENCES

1. 10 CFR 50.44.
2. 10 CFR 50, Appendix A.
3. FSAR Section 6.8.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 Isolation Valve Seal Water (IVSW) System

BASES

BACKGROUND

The Isolation Valve Seal Water (IVSW) System improves the effectiveness of certain containment isolation valves (CIVs) by providing a water seal to valve leakage paths. This is accomplished by injecting water between the seats and stem packing of globe and double-disk type isolation valves and into the piping between other closed containment isolation valves. IVSW sealing water is maintained in a seal water supply tank filled with water and pressurized with nitrogen. The IVSW System is actuated in conjunction with automatic initiation of containment isolation and is applied to CIVs in lines connected to the Reactor Coolant System or exposed to the containment atmosphere during an accident. The seal water is injected at a pressure of at least 47 psig which is > 1.1 times the calculated peak containment pressure (P_a). For those valves sealed by IVSW, the possibility of leakage from the Containment or Reactor Coolant System to the atmosphere outside containment is eliminated because leakage will be from the IVSW system into the Containment.

The containment is designed with an allowable leakage rate not to exceed 0.1% of the containment air weight per day. The maximum allowable leakage rate is used to evaluate offsite doses resulting from a DBA. Confirmation that the leakage rate is within limit is demonstrated by the performance of a Type A leakage rate test in accordance with the Containment Leakage Rate Testing Program as required by LCO 3.6.1, "Containment." During the performance of the Type A test, no credit is taken for the IVSW System in meeting the containment leakage rate criteria. As such, in the event of a DBA without an OPERABLE IVSW System, both the whole body and thyroid offsite doses would be within the guidelines specified in 10 CFR Part 100.

Although IVSW is not needed to maintain plant releases such that the whole body and thyroid offsite doses would be within the guidelines specified in 10 CFR Part 100 based on Type A leakage

(continued)

BASES

BACKGROUND (continued)

testing, Indian Point 3 elected to consider IVSW as a seal system as described in Reference 3. This election allows leakage through CIVs sealed by IVSW to be excluded when calculating Type B and C testing results. Therefore, operation of IVSW is an implicit assumption in the calculation of post accident offsite radiation doses.

To satisfy the requirements of Reference 3, for excluding leakage from CIVs sealed by IVSW from Type B and C limits, Technical Specifications must ensure the IVSW sealing function (i.e., both sealing water supply and nitrogen gas supply) is maintained at a pressure of 1.10 P_a for at least 30 days.

Sealing water design capacity is sufficient to maintain a source of seal water at the required pressure for a minimum of 24 hours without operator intervention assuming worst case leakage and the single failure of a CIV sealed by IVSW. The requirements for a 24 hour supply of seal water under worst case conditions is satisfied by maintaining a minimum of 144 gallons in the 176 gallon capacity seal water tank.

Nitrogen gas for IVSW seal water pressurization is satisfied by having three compressed nitrogen bottles in the IVSW supply bank aligned to the IVSW supply tank.

To satisfy the requirement of Reference 3 for maintaining the IVSW sealing function for at least 30 days, manual operator action may be required to replenish the IVSW seal water supply and/or compressed gas supply. Two sources of makeup water and two alternate sources of compressed gas with sufficient capacity to maintain the IVSW sealing function for 30 days are available. The two sources of makeup water are the primary water storage tank and the city water system. The two alternate sources of compressed gas are the normally isolated nitrogen gas bottles in the nitrogen supply bank and the ability to refill or replace the IVSW nitrogen supply bottles from the plant Nitrogen System. Manual operations required to supply makeup water and gas to the IVSW system are performed in an area that is

(continued)

BASES

BACKGROUND (continued)

accessible during an accident. The IVSW tank is instrumented to provide local indication of pressure and water level. Low water level, low pressure and high pressure in the IVSW supply tank are alarmed.

The IVSW System distribution piping consists of five headers. Three of the five IVSW headers are pressurized by opening either of a pair of normally closed air operated header injection valves. These valves open automatically on a containment Phase "A" isolation signal to admit seal water to the associated CIVs. One of the five IVSW headers is pressurized by opening either of a pair of normally closed, air-motor operated, header injection valves. These valves open automatically on a containment Phase "A" isolation signal to admit seal water to the associated CIVs. One IVSW header is used to supply seal water to CIVs on process lines that are not automatically closed on a containment Phase "A" isolation signal. This header is normally pressurized by the IVSW System with a normally closed manual or air-motor operated isolation valve for each pair of CIVs served by this IVSW header.

Redundant automatic header injection valves in parallel ensures the IVSW header is pressurized if there is a failure of one injection valve. Each of the two automatic header injection valves in each pair are actuated from separate and independent signals.

A related system, the Isolation Valve Seal Gas System, is not credited as a seal system as described in Reference 3, and is not addressed by this Technical Specification. This system uses the nitrogen bank used to supply the IVSW System to supply high pressure nitrogen that may be used to seal lines subjected to pressure in excess of the 150 psig IVSW design pressure due to operation of the recirculation pumps. This system is manually initiated during the post accident recovery phase and is not part of the IVSW System.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The IVSW System LCO was derived from the requirement related to the control of leakage from the containment during major accidents. This LCO is intended to ensure the actual containment leakage rate is maintained within the maximum value assumed in the safety analyses. As part of the containment boundary, containment isolation valves function to support the leak tightness of the containment. The IVSW System assures the effectiveness of certain containment isolation valves by providing a water seal pressurized to ≥ 1.1 times the maximum peak containment accident pressure at the valves and thereby reducing containment leakage. As such, the IVSW System is considered a seal system as described in Reference 3. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBA that results in a release of radioactive material within containment is a loss of coolant accident (LOCA)(Ref. 2). The DBAs assume that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L_a. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power) and containment isolation valve stroke time. The IVSW System actuates on a containment isolation signal and functions within 60 seconds to help reduce containment leakage within the allowable design leakage rate value, L_a.

The Isolation Valve Seal Water System satisfies Criterion 3 of 10 CFR 50.36.

LCO

OPERABILITY of the IVSW System is based on the system's capability to supply seal water to selective containment isolation valves within the time assumed in the applicable safety analyses and to ensure pressure is maintained for at least 30 days. This requires the IVSW tank be maintained with an adequate volume of water, an air or nitrogen overpressure sufficient to provide the motive force to move the water to the applicable

(continued)

BASES

LCO (continued)	penetration, piping to provide an OPERABLE flow path and two air operated header injection valves on each automatically actuated branch header.
APPLICABILITY	The IVSW System is required to be OPERABLE in MODES 1, 2, 3, and 4 because a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the IVSW System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

With one IVSW System header inoperable, a portion of the CIVs serviced by IVSW may not receive seal water at the required pressure and volume for effective sealing. However, the CIVs are OPERABLE and will still close, the affected CIVs provide adequate isolation to meet containment isolation requirements without IVSW during the most recent Type A test, and the number of CIVs affected by the failure of one IVSW header is small compared to the total number of CIVs. Therefore, the 7 days is allowed to restore the IVSW System header to OPERABLE status.

With one IVSW automatic actuation valve inoperable, the IVSW function is still available because the redundant automatic actuation valve is OPERABLE. Therefore, the 7 days is allowed to restore the IVSW automatic actuation valve to OPERABLE status.

B.1

With the IVSW system inoperable for reasons other than Condition A, the effectiveness of CIVs sealed by IVSW may be compromised. This Condition may result from failure to meet any of the surveillance requirements needed to verify Operability of IVSW or the inoperability of multiple IVSW headers or automatic actuation devices. However, the CIVs are OPERABLE and will still close and the affected CIVs provide adequate isolation to meet containment

(continued)

BASES

ACTIONS

B.1 (continued)

isolation requirements without IVSW during the most recent Type A test. Additionally, except in the unusual case where inoperability is the result of failure to meet SR 3.6.9.5, the affected CIVs have demonstrated the ability to satisfy IVSW leakage requirements using IVSW seal water in lieu of meeting Type C testing requirements. Therefore, the 24 hours is allowed to restore the IVSW System to OPERABLE status.

C.1 and C.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

This SR verifies the IVSW tank has the necessary pressure to provide motive force to the seal water. A 47 psig pressure is sufficient to ensure the containment penetration flowpaths that are sealed by the IVSW System are maintained at a pressure equal to or greater than 1.1 times the calculated peak containment internal pressure (Pa) related to the design bases accident. Verification of the IVSW tank pressure on a Frequency of once per 24 hours is acceptable because operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.9.2

This SR ensures the capability of the IVSW nitrogen source to pressurize the IVSW system as needed to support IVSW operation for a minimum of 30 days. Verification of the IVSW tank pressure on a Frequency of once per 24 hours is acceptable because operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.6.9.3

This SR verifies the IVSW tank has an initial volume of water necessary to provide seal water to the containment isolation valves served by the IVSW System for a period of at least 24 hours assuming the failure of one CIV and the maximum allowed leakage past other CIVs served by IVSW. Verification of IVSW tank level on a Frequency of once per 24 hours is acceptable since tank level is monitored by installed instrumentation and will alarm in the Primary Auxiliary Building prior to level decreasing to 20 gallons which provides sufficient time to re-fill the tank before it is depleted.

SR 3.6.9.4

This SR verifies the stroke time of each automatic IVSW header injection solenoid valve is within limits. The frequency is 24 months. Previous operating experience has shown that these valves usually pass the required test when performed.

SR 3.6.9.5

This SR ensures that automatic header injection valves actuate to the correct position on a simulated or actual signal. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.9.5 (continued)

Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.6.9.6

Integrity of the IVSW seal boundary is important in providing assurance that the design leakage value required for the system to perform its sealing function is not exceeded. This testing is performed in accordance with the requirements, Frequency and acceptance criteria established in Specification 5.5.15, Containment Leakage Rate Testing Program. This program was established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by IP3 specific approved exemptions. This program conforms to guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, dated September 1995."

REFERENCES

1. FSAR, Section 6.
2. FSAR, Chapter 14.
3. 10 CFR 50, Appendix J, Option A, Section III. B

B 3.6 CONTAINMENT SYSTEMS

3.6.10 Weld Channel and Penetration Pressurization System

BASES

BACKGROUND

The Weld Channel and Penetration Pressurization System (WC&PPS) is designed to continuously pressurize the double penetration barriers used at locations where plant systems penetrate the containment boundary, the space between selected isolation valves, and most of the weld seam channels installed on the inside of the liner of the Containment. Continuous pressurization by the WC&PPS provides a continuous, sensitive, and accurate means of monitoring their status with respect to leakage. Additionally, the WC&PPS is maintained at a pressure above the containment peak accident pressure so that any postulated leakage past the monitored barriers will be into the containment rather than out of the containment. The design basis leakage rate from the WC&PPS is 0.2% of containment free volume per day which assumes leakage of 0.1% of containment free volume per day into the containment and an identical amount leaking to the environment. Following a design basis accident, the system will maintain pressure greater than the post accident containment pressure for 24 hours (Ref. 1).

The WC&PPS is divided into four independent zones to simplify the process of locating leaks during operation. If one zone has a leak during operation, the specific penetration, weld channel, or containment isolation valve (CIV) containing the leak can be identified by isolating the individual air supply line to each component in the zone. Additionally, a capped tube connection installed in each line allows injecting leak test gas (Ref. 1).

The instrument air system provides a regulated supply of clean and dry compressed air for the WC&PPS. Two instrument air compressors are used, although only one is required to maintain pressurization at the maximum allowable leakage rate of the

(continued)

BASES

BACKGROUND (continued)

WC&PPS. A backup source of air for the WC&PPS is the station air system which includes at least one station air compressor. Each WC&PPS zone is served by its own air receiver which will continue to supply air to the zone if the instrument air system and station air system are lost. Each of the air receivers is sized to supply air to its zone for a period of at least one hour based on a total leakage rate of 0.2% of the containment free volume per day. If the receivers are exhausted before normal and backup air supplies are restored, additional backup is provided by a bank of nitrogen cylinders. The nitrogen backup system will automatically deliver nitrogen at a pressure slightly lower than the normal regulated air supply. Thus, in the event of failure of the normal and backup air supply systems during periods when the system is in operation, WC&PPS pressure requirements will be automatically maintained by the nitrogen supply. This assures reliable pressurization under both normal and accident conditions. The combination of the air receivers and nitrogen supply is sufficient to ensure WC&PPS pressure is above the peak containment pressure at the start of a LOCA and to maintain WC&PPS above the post-LOCA containment pressure profile for the 24 hour period following a LOCA at the design leakage rate of 0.2% of the containment free volume per day.

Pressure control valves, isolation valves and check valves are generally located outside of the containment for ease of inspection and maintenance. The line to each of the four pressurized zones is equipped with a critical pressure drop orifice to assure that air consumption will be within the capacity of the system and that high air consumption in one zone does not affect the operation of the other zones. Additionally, restricting orifices are installed on pressurization lines, where required, to assure that air consumption, even on failure of an individual line, will not result in loss of pressure to the other components connected to the same pressurization header.

All pressurized components have provisions for either local pressure indication, mounted outside the Containment, or remote low pressure alarms in the Control Room. The actuating pressure for each pressure alarm is set above incident pressure and below the nitrogen supply regulator setting.

(continued)

BASES

BACKGROUND (continued)

WC&PPS air consumption is continuously monitored by a flow sensing device located in each of the headers supplying makeup air to the four WC&PPS zones. Output from these sensors is applied to a summing amplifier which drives a total flow recorder. The flow measurement range is 0-15 scfm with an accuracy of ± 1% of full scale. High flow alarms in the Control Room are derived from the recording channel. With the WC&PPS at 43 psig and the containment at approximately atmospheric pressure, an indicated WC&PPS flow rate of 14.2 scfm is equivalent to the WC&PPS design leakage limit. A WC&PPS flow rate of 14.2 scfm, if sustained for 24 hours, is equivalent to 0.2% of the containment free volume at a pressure of 43 psig.

APPLICABLE SAFETY ANALYSES

For Indian Point 3, offsite dose calculations demonstrate compliance with 10 CFR 100 guidelines and the results are well within those guidelines. In these calculations, it is assumed that the Containment leaks at a rate of 0.1% per day of Containment free volume for the first 24 hours and 0.05% per day of Containment free volume thereafter. No credit is taken for the WC&PPS when determining the amount of radioactivity released for offsite dose evaluations because the integrated leakage rate tests required by Specification 5.15, Containment Leakage Rate Testing Program, are performed with the double penetration and weld channel zones open to the containment atmosphere. However, WC&PPS does provide an additional means for ensuring that containment leakage is minimized (Ref. 3).

A design function of WC&PPS is to provide a continuous, sensitive, and accurate means of monitoring leakage of selected containment isolation valves (CIVs), the air lock door seals, and containment welds that are pressurized by this system. WC&PPS leakage, even if below the WC&PPS design leakage rate, may indicate that one of these supported components is exceeding its leakage rate acceptance criteria. In this situation, the supported component may be inoperable and the APPLICABLE SAFETY ANALYSES for the supported component is applicable.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Specification 5.15, Containment Leakage Rate Testing Program, allows an exemption to Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, and ANS 56.8-1994, Section 3.3.1, in that WC&PPS supply isolation valves are not required to be Type C tested. Note that the WC&PPS supply isolation valves are normally open valves. As specified in Reference 2, operating with these valves normally open and the exemption from type C testing is acceptable because: (1) the WC&PPS is monitored for changes to the system leakage rate; (2) the WC&PPS leakage rate is quantified every 36 months; and, (3) WC&PPS pressure is maintained higher than the containment peak accident pressure (Ref. 2). Therefore, if the required pressure is not maintained or excessive WC&PPS leakage is identified, then compensatory actions are required to ensure the containment boundary is maintained.

<u>For containment isolation valves (CIVs) supported by WC&PPS</u>, WC&PPS pressurization is applied to the space between those CIVs that are normally closed. CIVs supported by WC&PPS are Type C tested in accordance with Specification 5.5.15 because WC&PPS is not credited as a seal system. <u>For loss of WC&PPS pressurization</u>, isolation of the WC&PPS supply to the affected CIVs provides appropriate compensatory action because the supported CIVs are a tested boundary and isolating the depressurized WC&PPS supply eliminates WC&PPS as a potential leakage path. <u>For high WC&PPS air consumption</u>, a consideration is that the leakage may indicate that a supported CIV is exceeding its leakage rate acceptance criteria. If the leakage path is isolated from the supported CIVs when the WC&PPS supply to the CIV is isolated, isolation of the WC&PPS supply to the CIV restores the required safety function. If the leakage path is not isolated from the supported CIV when the WC&PPS supply to the CIV is isolated (i.e., the CIV is depressurized), the supported CIV may be inoperable and the requirements of LCO 3.6.3, "Containment Isolation Valves," are applicable.

<u>For the containment air lock door seals supported by WC&PPS</u>, WC&PPS pressurization is normally applied to the space between the double gaskets on each of the airlock seals.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Air lock operability does not require pressurization of the air lock door seals except as needed to verify the seals have reseated after each air lock door is operated (see LCO 3.6.2, "Containment Air Locks"). For loss of WC&PPS pressurization, isolation of the WC&PPS supply to the affected air lock door seals provides appropriate compensatory action because pressurization is not required for air lock operability (except as needed to verify the seals have reseated after each air lock door is operated) and isolating the depressurized WC&PPS supply eliminates WC&PPS as a potential leakage path. For high WC&PPS air consumption, a consideration is that the leakage may indicate that a supported air lock seal is exceeding its leakage rate acceptance criteria. If the leakage path is isolated from the supported air lock when the WC&PPS supply to the air lock is isolated, isolation of the WC&PPS supply to the air lock restores the required safety function. If the leakage path is not isolated from the supported air lock seal when the WC&PPS supply to the air lock seal is isolated, the supported air lock may be inoperable and the requirements of LCO 3.6.2, "Containment Air Locks," are applicable.

For weld channels and piping penetrations supported by WC&PPS, WC&PPS pressurizes what is equivalent to a closed system inside containment. Because it is reasonable to assume that WC&PPS leakage is not the result of a containment weld or piping penetration defect, WC&PPS leakage and/or lack of pressurization is a concern only because it presents a potential leakage path from containment to the atmosphere via the depressurized WC&PPS. Therefore, isolation of the WC&PPS supply to the affected section of weld channel or piping penetration provides appropriate compensatory action for both loss of pressurization and air consumption caused by flow from the WC&PPS into containment. This assumes that containment leakage rate testing required by Specification 5.15 provides a high degree of assurance that WC&PPS air consumption is not indicative of deterioration of the containment boundary.

WC&PPS satisfies Criterion 3 of 10 CFR 50.36 where it is used to pressurize the space between selected CIVs and pressurize air

(continued)

APPLICABLE SAFETY ANALYSES (continued)

lock door seals. The WC&PPS system, if not maintained at the required pressure, represents a potential leakage path to the environment if there is a single failure of a supported CIV or air lock seal.

WC&PPS satisfies Criterion 4 of 10 CFR 50.36 it provides an additional means for ensuring that containment leakage is minimized although no credit is taken for the WC&PPS in calculating offsite dose for meeting 10 CFR 100 and GDC 19.

LCO

This LCO requires that the WC&PPS be OPERABLE. OPERABILITY requires the following: all required portions of each WC&PPS zone are pressurized to a value that exceeds peak containment pressure during a design basis accident; and, total leakage (i.e., air consumption) from the required portions of the WC&PPS are within specified limits. Limits for air consumption are based on the integrated containment leak rate test acceptance criterion and the ability of the reserve air supplies in the air receivers and nitrogen cylinders to maintain WC&PPS pressure above calculated containment pressure for a minimum of 24 hours following an event.

For a portion of the WC&PPS to be considered not required, it must meet all of the following criteria: 1) it must be inoperable (i.e., can not maintain a pressure above required limits and/or cause system air consumption to exceed required limits); 2) it must be isolated or disconnected from the system; and, 3) it must have been determined by written evaluation as not practicably accessible for repair.

Inoperable sections of WC&PPS piping which can be considered as not practicably accessible for repair will satisfy one of the following criteria: 1) the piping is covered by concrete and repairs of the piping would involve the removal of some portion of the containment structure; or, 2) the piping is located behind plant equipment in the containment building and repairs of the piping would involve the relocation of the equipment.

(continued)

BASES

LCO (continued)

The integrity of the welds associated with any disconnected or isolated portions of the WC&PPS is considered verified by integrated leak rate testing performed in accordance with Specification 5.15. The provision that allows for the disconnection of portions of the WC&PPS piping does not apply to any other WC&PPS piping.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. WC&PPS is required to support OPERABILITY of the containment, containment air locks, and selected containment isolation valves. In MODES 5 and 6, OPERABILITY of the containment, containment air locks, and containment isolation valves is not required. Therefore, the WC&PPS is not required to be OPERABLE in MODES 5 and 6.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 is added to clarify that Separate Condition entry is allowed for each component supplied by WC&PPS. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each component supported by WC&PPS. Complying with the Required Actions may allow for continued operation, and subsequent inoperable WC&PPS components are governed by subsequent Condition entry and application of associated Required Actions.

Note 2 is added to direct entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment," if it is determined that WC&PPS inoperability is indicative of exceeding the overall containment leakage rate. Note that entry into the Conditions and Required Actions of LCO 3.6.1 may be required even if WC&PPS air consumption limits are not exceeded.

A.1 and A.2

In the event one or more components supplied by WC&PPS is not within the pressure limit of SR 3.6.10.1, Required Action A.1 requires that the WC&PPS supply to the affected weld channels, penetrations, or containment isolation valves must be isolated within 4 hours. Required Action A.1 is needed because isolation

(continued)

ACTIONS

A.1 and A.2 (continued)

of the WC&PPS supply to the affected component results in using an isolation valve as a substitute for pressurization. This prevents the WC&PPS from becoming a potential leakage path from the containment to the atmosphere. This action satisfies the required safety function because the leakage rate testing performed in accordance with Specification 5.15 has already verified that the containment leakage rate is within required limits without crediting the WC&PPS.

The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, a blind flange (including Swagelok fittings), and a check valve with flow through the valve secured (Ref. 3). For a WC&PPS supply isolated in accordance with Required Action A.1, the device used to isolate the weld channel, penetration or containment isolation valves should be the closest available to component. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

If a WC&PPS supply cannot be restored to OPERABLE status within the 4 hour Completion Time and is isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and not pressurized by WC&PPS will be in the isolation position should an event occur. Required Action A.2 does not require any testing or device manipulation. This action involves verification, through a system walkdown, that isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" and exempting valves that are locked, sealed or otherwise secured in the required position is appropriate considering the fact that the devices are operated under administrative controls and the

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

B.1, B.2 and B.3

Condition B applies if WC&PPS has air consumption that places the WC&PPS outside the limits of SR 3.6.10.2. In this condition, Required Action B.3 requires that portions of the WC&PPS are isolated, as necessary, to restore WC&PPS leakage to within the limits of SR 3.6.10.2. However, safety function is not restored until any portions of the WC&PPS that are depressurized by this Action are isolated. Therefore, Required Action B.3, is modified by a Note that requires entry into Condition A for components not within the pressure limit of SR 3.6.10.1 as a result of isolating the leakage path. The Completion Time of 7 days to isolate the leakage path is acceptable because all un-isolated portions of the WC&PPS are pressurized, otherwise, Condition A is applicable immediately. Safety function is restored when leaking portions of the WC&PPS are isolated and at least one isolation device separates the containment barrier from the WC&PPS leakage path. If leakage exceeds 0.2%, then replenishment would be required before 24 hours, during an accident.

(continued)

BASES

ACTIONS

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

As discussed in the Applicable Safety Analyses above, safety function is not restored by Required Action B.3 if the air consumption leakage path is depressurized but not isolated from the supported containment isolation valves or containment air lock seal. In this situation, the WC&PPS air consumption leakage path could create a leakage path from containment to the atmosphere. Therefore, Required Action B.1 requires entry into the applicable Conditions and Required Actions of LCO 3.6.3, "Containment Isolation Valves" within 1 hour of discovery that the WC&PPS air consumption leakage path is depressurized and not isolated from the supported containment isolation valves. Likewise, Required Action B.2 requires entry into the applicable Conditions and Required Actions of LCO 3.6.2, "Containment Air Locks" within 1 hour of discovery that the WC&PPS air consumption leakage path is depressurized and not isolated from the supported air locks. The Required Actions of LCO 3.6.2 and LCO 3.6.3 will restore safety function for WC&PPS air consumption leakage path that is depressurized.

C.1 and C.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1

This SR requires periodic verification during plant operation that the required portions of each WC& PPS zone are maintained at a pressure greater than the containment peak accident pressure.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1 (continued)

This SR is satisfied by verification of zone pressure on each of the four WC&PPS zones is above the specified limit. The 31 day Frequency is acceptable because there are low pressure alarms in the Control Room to ensure that operators are aware that all WC&PPS zones are pressurized.

SR 3.6.10.2

This SR requires periodic verification during plant operation that the WC&PPS air consumption is ≤ 0.2% of the containment free volume per day. This SR is performed by taking the sum of the reading on the flow sensing devices located in each of the zone headers. A WC&PPS flow rate of 14.2 scfm, if sustained for 24 hours, is equivalent to 0.2% of the containment free volume at a pressure of 43 psig. The 31 day Frequency recognizes that WC&PPS air consumption indication and high flow alarms are provided in the control room.

SR 3.6.10.3

This SR, sometimes called the sensitive leak rate test, ensures that the leakage rate for the WC&PPS is ≤ 0.2% of the containment free volume per day when pressurized to ≥ 43 psig above containment pressure. The sensitive leak rate test includes only the volume of the weld channels, double penetrations, and containment isolation valves supported by WC&PPS. This test is considered more sensitive than the integrated leakage rate test, as the instrumentation used permits a direct measurement of leakage from the pressurized zones. The 36 month Frequency is acceptable because experience has shown that the WC&PPS usually passes this Surveillance when performed at the 36 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. The Frequency is modified by a Note indicating that SR 3.0.2 is not applicable.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 6.6.
2. Safety Evaluation Report for IP3 Amendment 174.
3. FSAR, Section 14.3.
4. Standard Review Plan Section 6.2.4.

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves and non-return valves, as described in the FSAR, Section 10.2 (Ref. 1). The five code safety valves per steam generator consist of four 6 inch by 10 inch and one 6 inch by 8 in. These valves are set to open at 1065, 1080, 1095, 1110 and 1120 psig, respectively. The steam generator safety valve capacity is rated to remove the maximum calculated steam flow (normally 105% of the maximum guaranteed steam flow) from the steam generators without exceeding 110% of the steam system design pressure, (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine or reactor trip.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to $\leq$ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section 14 (Ref. 3). Of these, the full power loss of external electrical load without steam dump is the limiting AOO.

The transient response for loss of external electrical load without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. If a minimum reactivity feedback is assumed, the reactor is tripped on high pressurizer pressure. In this case, the pressurizer safety valves open, and RCS pressure remains below 110% of the design value. The MSSVs also open to limit the secondary steam pressure.

If maximum reactivity feedback is assumed, the reactor is tripped on overtemperature ΔT. The departure from nucleate boiling ratio increases throughout the transient, and never drops below its initial value. Pressurizer relief valves and MSSVs are activated and prevent overpressurization in the primary and secondary systems.

Startup and power operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. This is accomplished by reducing the neutron flux trip setpoint and reducing THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. These limits on the neutron flux trip setpoint, specified in Table 3.7.1-1, are established based on guidance provided in Nuclear Safety Advisory Letter (NSAL) 94-001, Operation at Reduced Power Levels with Inoperable Main Steam Safety Valves (Ref. 6) and Information Notice 94-60, Potential Overpressurization of Main Steam System (Ref. 7). The reactor trip setpoint reductions are calculated as follows:

$$Hi\phi = (100 / Q) [(wsh_{fg}N) / K]$$

Where:

$Hi\phi$ = Safety Analysis high neutron flux setpoint (% RTP);

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat) in Mwt (i.e.,3037 Mwt);

K = Conversion factor, 947.82 (Btu/sec)/Mwt;

ws = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure, including tolerance and accumulation, as appropriate, in lb/sec. (ws = 150 + 228.61 * (4 - V) lb/sec, where V = Number of inoperable safety valves in the steam line of the most limiting steam generator).

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm (i.e.,608.5 Btu/lbm).

N = Number of loops in plant (i.e., 4).

The calculated reactor trip setpoint is further reduced by 9% of full scale to account for instrument uncertainty and then rounded down.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36.

LCO

The accident analysis requires five MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 102% RTP. An MSSV will be considered inoperable if it fails to open on demand. The LCO requires that five MSSVs be OPERABLE in compliance with Reference 2. This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements). These limitations are according to Table 3.7.1-1 in the accompanying LCO, and Required Action A.1.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced.

(continued)

BASES

LCO (continued)

The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB.

APPLICABILITY

In MODE 1 above 23% RTP, the number of MSSVs per steam generator required to be OPERABLE must be according to Table 3.7.1-1 in the accompanying LCO. Below 23% RTP in MODES 1, 2, and 3, only two MSSVs per steam generator are required to be OPERABLE.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1

Startup and power operation with up to three of the five MSSVs associated with each steam generator inoperable is permissible if the maximum allowed power level is below the heat removing capability of the operable MSSVs. Therefore, startup and power operation with inoperable main steam line safety valves is allowable if the neutron flux trip setpoints are restricted within the limits specified in Table 3.7.1-1. This ensures that reactor power level is limited so that the heat input from the primary side will not exceed the heat removing capability of the OPERABLE MSSVs of the most limiting steam generator.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status within the associated Completion Time, or if one or more steam generators have less than two MSSVs OPERABLE, 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 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR, Section 10.2.
2. ASME, Boiler and Pressure Vessel Code, Section III, 1971 Edition.
3. FSAR, Section 14.
4. ASME, Boiler and Pressure Vessel Code, Section XI.
5. ANSI/ASME OM-1-1987.
6. Nuclear Safety Advisory Letter (NSAL) 94-001, Operation at Reduced Power Levels with Inoperable Main Steam Safety Valves.
7. Information Notice 94-60, Potential Overpressurization of Main Steam System.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs) and Main Steam Check Valves (MSCVs)

BASES

BACKGROUND

The Main Steam System conducts steam from each of the four steam generators within the containment building to the turbine stop and control valves. The four steam lines are interconnected near the turbine. Each steam line is equipped with an isolation valve identified as the Main Steam Isolation Valve (MSIV) and a non-return valve identified as the Main Steam Check Valve (MSCV).

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators.

The MSIVs are swing disc type check valves that are aligned to prevent flow out of the steam generator. During normal operation, the free swinging discs in the MSIVs are held out of the main steam flow path by an air piston and the MSIVs close to prevent the release of steam from the SG when air is removed from the piston. The isolation valves are designed to and required to close in less than five seconds. The MSIV operators are supplied by instrument air and each MSIV is equipped with an air receiver to prevent spurious MSIV closure due to pressure transients in the instrument air system.

Each MSIV is equipped with a bypass valve used to warm up the steam line during unit startup which equalizes pressure across the valve allowing it to be opened. The bypass valves are manually operated and are closed during normal plant operation.

An MSIV closure signal is generated by the following signals:

High steam flow in any two out of the four steam lines coincident with low steam line pressure; or,

High steam flow in any two out of the four steam lines coincident with low Tavg; or,

(continued)

BASES

BACKGROUND (continued)

Two sets of the two-of-three high-high containment pressure signals; or,

Manual actuation using a separate switch in the control room for each MSIV.

Note that a turbine trip is initiated whenever an MSIV is not fully open.

The MSCVs are swing disc type check valves that are aligned to prevent reverse flow of steam into an SG if an individual SG pressure falls below steamline pressure.

One MSIV and one MSCV are located in each main steam line outside but close to containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, Steam Bypass System (High Pressure Steam Dump), and other auxiliary steam supplies from the steam generators.

A description of the MSIVs and MSCVs is found in the FSAR, Section 10.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment (Ref. 2) and the accident analysis of the SLB events presented in the FSAR, Sections 6.2 and 14.2 (References 2 and 3, respectively). The combination of MSIVs and MSCVs precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand). For a break upstream of an MSIV, either the MSIVs in the other three steam lines or the MSCV in the steam line with the faulted SG must close to prevent the blowdown of more than one SG. For a break downstream of an MSIV, the MSCVs are not required to function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting case for the containment analysis is the SLB inside containment, without a loss of offsite power and failure to close of the MSCV on the affected steam generator or the failure to close of the MSIV associated with any other SG. With either of these failures, only one SG blows down.

The limiting SLBs occur at low power or hot shutdown because the magnitude and duration of the RCS cooldown will be greater if the SLB is initiated from these conditions. This occurs because, at low power conditions, there is less stored energy in the fuel and the initial steam generator water inventory is greatest at no load. Additionally, the magnitude and duration of the RCS cooldown will be greater if RCPs continue to operate during the SLB. Therefore, an SLB without loss of offsite power is more limiting.

If it is assumed that the most reactive rod cluster control assembly is stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. In the most limiting condition, the core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at hot zero power with offsite power available is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available, and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System cooldown. With a loss of offsite power, the response of mitigating systems is delayed.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Significant single failures considered include: 1) failure of an MSIV or MSCV to close; 2) failure of a feedwater control or isolation valve to close; 3) failure of a diesel generator; and, 4) failure of auxiliary feedwater pump runout protection.

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. A HELB inside containment. In order to maximize the mass and energy release into containment, the analysis assumes that the MSCV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from all steam generators until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops. Closure of the MSIVs isolates the break from the unaffected steam generators.

b. A break outside of containment and upstream from the MSIVs. This case is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs. This case will be isolated by the closure of the MSIVs.

d. Following a steam generator tube rupture. In this case, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36.

LCO

This LCO requires that four MSIVs and four MSCVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal. The MSCVs are considered OPERABLE when inspections and testing required by the Inservice Test Program are completed at the specified FREQUENCY in accordance with SR 3.7.2.2.

This LCO provides assurance that the MSIVs and MSCVs 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 and MSCVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when MSIVs are closed. These are the conditions when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low and the potential for and consequences of an SLB are significantly reduced.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

(continued)

BASES (continued)

ACTIONS

A.1

With one or more MSCVs inoperable, action must be taken to restore OPERABLE status within 48 hours. In this condition, the MSIVs in the other three steam lines must close to prevent the blowdown of more than one SG following an SLB upstream of an MSIV. Having more than one MSCV inoperable will not increase the consequences of an SLB upstream of an MSIV because only the MSCV associated with the faulted SG needs to function to mitigate the failure of an MSIV associated with any of the other SGs. Additionally, an inoperable MSCV does not affect the consequences of an SLB downstream of the MSIV.

The 48 hour Completion Time is acceptable because of the following: all MSIVs are Operable, there is a low probability of the failure of an MSIV during the 48 hour period that one or more MSCVs are inoperable; and, there is a low probability of an accident that would require a closure of the MSCVs or MSIVs during this period.

B.1, B.2 and B.3

If the MSCVs cannot be restored to OPERABLE status within 48 hours, 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 MODE 2 within 6 hours and all MSIVs must be closed within 14 hours. The Completion Times are reasonable, based on operating experience, to reach MODE 2 and to close the MSIVs or complete a plant cooldown to MODE 4 in an orderly manner and without challenging unit systems.

If an inoperable MSCVs cannot be restored to OPERABLE status within the specified Completion Time, then all MSIVs must be verified to be closed on a periodic basis while the plant is in MODE 2 or 3. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

(continued)

BASES

ACTIONS (continued)

C.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 48 hours. Some repairs to the MSIV can be made with the unit hot. The 48 hour Completion Time is acceptable because the four OPERABLE MSCVs prevent the blowdown of more than one SG following an SLB upstream of the MSIV even if more than one MSIV fails to close. Additionally, there is a low probability of the failure of an MSCV during the 48 hour period that the MSIV is inoperable; and, there is a low probability of an accident that would require a closure of the MSIVs occurring during this time period.

The 48 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from most other containment isolation valves in that the closed system provides an additional means for containment isolation.

D.1

If the MSIV cannot be restored to OPERABLE status within 48 hours, 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 MODE 2 within 6 hours and Condition E would be entered. The Completion Times are reasonable, based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging unit systems.

E.1 and E.2

Condition E is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The 8 hour Completion Time is reasonable, based on operating experience, to close the MSIVs after reaching MODE 2 or complete a plant cooldown to MODE 4 in an orderly manner and without challenging unit systems.

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, the inoperable MSIVs must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

F.1 and F.2

If one MSIV is inoperable when one or more MSCVs are inoperable, then more than one SG may blowdown following an SLB upstream of an MSIV and the plant is outside of the analysis assumptions. The plant remains within the analysis assumptions for an SLB downstream of an MSIV although the ability to tolerate the failure of a second MSIV is lost. In this condition, all MSCVs must be restored to OPERABLE status or all MSIVs must be restored to OPERABLE status within 8 hours.

The 8 hour Completion Time is acceptable because of the low probability of an accident that would require a closure of the MSCVs or MSIVs during this time period. The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a

(continued)

BASES

ACTIONS

F.1 and F.2 (continued)

closed system penetrating containment. These valves differ from most other containment isolation valves in that the closed system provides an additional means for containment isolation.

G.1 and G.2

If the MSIVs or MSCVs cannot be restored to OPERABLE status or are not closed 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 at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 5.0 seconds on an actual or simulated actuation signal. The MSIV closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs are not tested at power because even a part stroke causes a turbine trip and valve closure. As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program. The Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at this Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

This test is conducted in MODE 3 with the unit at operating temperature and pressure, as discussed in Reference 5. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

Each MSCV must be inspected to ensure that it closes properly. This ensures that the safety analysis assumptions are met. The Frequency of this SR is based on Inservice Testing Program requirements and corresponds to the expected refueling cycle.

REFERENCES

1. FSAR, Section 10.2.
2. FSAR, Section 6.
3. FSAR, Section 14.
4. 10 CFR 100.11.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (MFRVs) and MFRV Low Flow Bypass Valves

BASES

BACKGROUND

The MBFPDVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). The safety related function of the MFRVs is to provide the second isolation of MFW flow to the secondary side of the steam generators following an HELB. Closure of the two MBFPDVs or four MFRVs and four MFRV low flow bypass valves terminates flow to the steam generators. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MBFPDVs will be mitigated by their closure. Closure of the MBFPDVs or MFRVs and MFRV low flow bypass valves, effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs.

In the event of a secondary side pipe rupture inside containment, either the MBFPDVs or MFRVs and MFRV low flow bypass valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops.

One MBFPDV is located on the discharge of each of the two Main Boiler Feedpumps (MBFPs), and one MFRV and MFRV low flow bypass valve, is located on each of the four MFW lines, outside but close to containment. The MFIVs and MFRVs are located upstream of the AFW injection point so that AFW may be supplied to the steam generators following MBFPDV or MFRV closure. The piping volume from these valves to the steam generators must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

(continued)

BASES

BACKGROUND (continued)

The two MBFPDVs, four MFRVs and four MFRV low flow bypass valves will close on receipt of an ESFAS Safety Injection signal. An ESFAS Tavg – Low coincident with reactor trip will close the four MFRVs and four MFRV low flow bypass valves. A Steam Generator Hi-Hi level trip will close the MBFPDV and MFRVs and MFRV low flow bypass valves associated with the affected SG. They may also be closed manually. In addition to the two MBFPDVs, four MFRVs and four MFRV low flow bypass valves, a check valve outside containment is available. The check valve isolates the feedwater line to prevent blowdown of a SG if main or auxiliary feedwater pressure are lost.

A description of the MBFPDVs and MFRVs is found in the FSAR, Section 10.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MBFPDVs and MFRVs is established by the analyses for the large SLB. Closure of the MBFPDVs, MFRVs and MFRV low flow bypass valves, may also be relied on to terminate an SLB for core response analysis and excess feedwater event upon the receipt of a steam generator water level – high high or a feedwater isolation signal. Feedwater isolation also occurs as a result of any safety injection signal. Failure of an MBFPDV in conjunction with the failure of an MFRV or MFRV low flow bypass valve to close following an SLB can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

The MBFPDVs, MFRVs and MFRVs Low Flow Bypass Valves satisfy Criterion 3 of 10 CFR 50.36.

LCO

This LCO ensures that the MBFPDVs, MFRVs and MFRV low flow bypass valves will isolate MFW flow to the steam generators, following a main steam line break.

(continued)

BASES

LCO (continued)	This LCO requires that two MBFPDVs, four MFRVs and four MFRV low flow bypass valves be OPERABLE. The MBFPDVs, MFRVs and MFRV low flow bypass valves are considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal. Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. A feedwater isolation signal on a steam generator water level - high high signal and this function is relied on to terminate an excess feedwater flow event; therefore, failure to meet the LCO may result in the introduction of water into the main steam lines.
APPLICABILITY	The MBFPDVs, MFRVs and MFRV bypass valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, and 3, the MBFPDVs, MFRVs and MFRV bypass valves are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve, they are already performing their safety function. A de-activated motor operated valve is considered to be a manual valve. In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MBFPDVs, MFRVs and MFRV bypass valves are normally closed since MFW is not required.
ACTIONS	The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve. <u>A.1 and A.2</u> With one MFPDV in one or both flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves, the MBFP trip function, and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on industry operating experience.

Inoperable MBFPDVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

B.1 and B.2

With one MFRV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on industry operating experience.

Inoperable MFRVs, that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of other administrative controls to ensure that the valves are closed or isolated.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

With one MFRV low flow bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on industry operating experience.

Inoperable associated bypass valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of the administrative controls that ensure that these valves are closed or isolated.

D.1

With two inoperable valves in series in the same flow path, there may be no redundant system to operate automatically and perform the required safety function. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MBFPDV or MFRV, or otherwise isolate the affected flow path.

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

If the MBFPDV(s), MFRV(s), and MFRV bypass valve(s) cannot be restored to OPERABLE status, or closed, or isolated 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 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MBFPDV(s), MFRV(s), and MFRV bypass valves is within required limits on an actual or simulated actuation signal. The closure times are assumed in the accident and containment analyses. The acceptance criteria for this SR do not include the 2 second delay associated with the ESFAS activation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves can not be tested at power because valve closure or even a part stroke exercise increases the risk of a valve closure and MBFP trip. This is consistent with the ASME Code, Section XI (Ref. 2), quarterly stroke requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the Inservice Testing Program. The required Frequency for valve closure is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the required Frequency.

REFERENCES

1. FSAR, Section 10.2.

2. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

The ADVs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Bypass System (High Pressure Steam Dump) to the condenser not be available, as discussed in the FSAR, Section 10.2 (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST). The ADVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the High Pressure Steam Dump System.

One ADV line for each of the four steam generators is provided. Each ADV line consists of one ADV and an associated manually operated block valve.

The block valves are upstream of the ADVs to permit testing and maintenance at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

The ADVs are provided with a pressurized gas supply of bottled nitrogen that is needed to support manual operation of the atmospheric dump valves. The nitrogen supply is sized to provide the sufficient pressurized gas to operate the ADVs for the time required for Reactor Coolant System cooldown to RHR entry conditions.

A description of the ADVs is found in Reference 1.

APPLICABLE SAFETY ANALYSES

The design basis of the ADVs is established by the capability to cool the unit to RHR entry conditions. The total relief capacity of the four ADVs is approximately 10% of the rated steam flow.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

This is adequate to cool the unit to RHR entry conditions with only one steam generator and one ADV, utilizing the cooling water supply available in the CST.

In the accident analysis presented in Reference 1, the ADVs 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. Prior to operator actions to cool down the unit, the main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event, the operator is also required to perform a limited cooldown 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 the ADVs. The requirement that 3 of the 4 ADVs must be OPERABLE is established to ensure that at least one ADV line is available under local control to conduct a plant cooldown following an event in which one steam generator becomes unavailable due to the event (i.e., SGTR or SLB), accompanied by a single, active failure of a second ADV line on an unaffected steam generator.

The ADVs are equipped with block valves in the event an ADV spuriously fails open or fails to close during use.

The ADVs satisfy Criterion 3 of 10 CFR 50.36.

LCO

Three of the four ADV lines are required to be OPERABLE. One ADV line is required from each of three steam generators to ensure that at least one ADV line is available to conduct a unit cooldown following an SGTR, in which one steam generator becomes unavailable, accompanied by a single, active failure of a second

(continued)

BASES

LCO (continued)

ADV line on an unaffected steam generator. The block valves must be OPERABLE to isolate a failed open ADV line. A closed block valve does not render it or its ADV line inoperable because operator action time to open the block valve is supported in the accident analysis.

Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an event in which the condenser is unavailable for use with the Turbine Steam Bypass System (High Pressure Steam Dump).

An ADV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand (either remotely or under local control).

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal, the ADVs are required to be OPERABLE.

In MODE 5 or 6, an SGTR is not a credible event.

ACTIONS

A.1

With one required ADV line inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE ADV lines. Specifically, with one of the three required ADVs inoperable, at least one ADV line is available to conduct a plant cooldown following an event in which one steam generator becomes unavailable due to the event (i.e., SGTR or SLB), accompanied by a single, active failure of a second ADV line on an unaffected steam generator. Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply.

(continued)

BASES

ACTIONS (continued)

B.1

With two or more required ADV lines inoperable, action must be taken to restore all but one ADV line to OPERABLE status. Since the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Bypass System (HP Steam Dump) and MSSVs, and the low probability of an event occurring during this period that would require the ADV lines.

C.1 and C.2

If the ADV lines cannot be restored to OPERABLE status 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 4, without reliance upon steam generator for heat removal, within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the ADVs must be able to be opened either remotely or locally and throttled through their full range. This SR ensures that the ADVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ADV during a unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the specified Frequency and, therefore, is acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.4.2

The function of the block valve is to isolate a failed open ADV. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the specified Frequency and, therefore, is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 10.2.

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction from the condensate storage tank (CST) (LCO 3.7.6) and pump to the steam generator secondary side via a connection to the main feedwater (MFW) piping at a point outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or atmospheric dump valves (LCO 3.7.4). If the main condenser is available, steam may be released via the steam bypass (High Pressure Steam Dump) valves and recirculated to the CST.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. FSAR Section 10.2 (Ref. 1) describes this configuration as two pumping loops using two different types of motive power to the pumps. One auxiliary feedwater loop utilizes a steam turbine driven pump and the other utilizes two motor driven pumps. Technical specifications describe this configuration as three trains because each motor driven pump provides 100% of AFW flow capacity, and, depending on steam conditions, the turbine driven pump capacity approaches 200% of the required capacity to the steam generators, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent power supply and feeds two steam generators. The steam turbine driven AFW pump receives steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump.

(continued)

BASES

BACKGROUND (continued)

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

The turbine driven AFW pump supplies a common header capable of feeding all steam generators. Each of the steam generators can also be supplied by one of the two motor driven AFW pumps. Any of the three pumps at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the ADVs.

The motor driven pumps are actuated by any one of the following:

1) Low-low level in any steam generator;

2) Loss of voltage (Non SI blackout) on 480 VAC bus 2A/3A (starts AFW Pump 31) and loss of voltage (Non SI blackout) on 480 VAC bus 6A (starts AFW Pump 33);

3) Safety Injection signal;

4) Auto trip of either main boiler feed pump;

5) Manual actuation from the Control Room; and

6) Manual actuation locally at the pump room.

The steam turbine driven pump is actuated by any one of the following:

1) Low-low level in two of the four steam generators;

2) Loss of voltage (Non SI blackout) on 480 VAC busses 2A/3A or 6A;

(continued)

BASES

BACKGROUND (continued)

3) Manual actuation from the Control Room; and

4) Manual actuation locally at the pump room.

The steam driven AFW pump must be throttled manually in order to bring the unit up to speed after a start signal. In addition, the steam driven pump discharge flow control valves must be manually opened as necessary to provide adequate auxiliary feedwater flow.

The AFW System is discussed in the FSAR, Section 10.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus accumulation.

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW System flow must also be available to account for flow losses such as pump recirculation and line breaks.

The limiting events that require the AFW System are as follows:

a. small break loss of coolant accident;

b. loss of AC sources; and

c. loss of feedwater.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The AFW turbine driven pump actuates automatically when required to ensure an adequate feedwater supply to the steam generators is available during loss of power. Power operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36.

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of events that could result in overpressurization of the reactor coolant pressure boundary. Three independent AFW pumps are required to be OPERABLE to ensure the capability to maintain the plant in hot shutdown with a loss of offsite power and a single failure. This is accomplished by powering two of the pumps from independent emergency buses. The third AFW pump is powered by a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs.

The AFW System is configured into three trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE, each supplying AFW to two separate steam generators. The turbine driven AFW pump is required to be OPERABLE with steam supplies from each of two main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to all of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE.

The LCO is modified by a Note indicating that one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. The motor driven AFW pump required to be OPERABLE in Mode 4 must be capable of supporting the SG(s) being credited as the redundant decay heat removal path in accordance with LCO 3.4.6, RCS Loops - MODE 4. This requirement ensures the ability to

(continued)

BASES

LCO (continued)	maintain the required level in the SG(s) (and decay heat removal capacity) during extended periods in Mode 4 with or without offsite power. Requiring only one OPERABLE AFW pump is acceptable because of the reduced heat removal requirements and short period of time in MODE 4 during which the AFW is required and the insufficient steam available in MODE 4 to power the turbine driven AFW pump.
APPLICABILITY	In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the MFW is lost. In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory needed to achieve and maintain MODE 4 conditions. In MODE 4, a motor driven AFW pump may be needed to support heat removal via the steam generators. In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.
ACTIONS	<u>A.1</u> If one of the two steam supplies to the turbine driven AFW train is inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons: a. The redundant OPERABLE steam supply to the turbine driven AFW pump; b. The availability of redundant OPERABLE motor driven AFW pumps; and c. The low probability of an event occurring that requires the inoperable steam supply to the turbine driven AFW pump.

(continued)

BASES

ACTIONS

A.1 (continued)

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

When Required Action A.1 or B.1 cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODE 1, 2, or 3, 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 4 within 18 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.

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

D.1

If all three AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

(continued)

BASES

ACTIONS (continued)

E.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With one required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a Note that states the SR is not required in MODE 4. Not performing this SR in MODE 4 is acceptable for the following reasons: AFW pumps are typically operated intermittently to keep the SGs filled when in MODE 4, the decay heat load is low; an RHR loop is required to be OPERABLE as the primary method of decay heat removal in Mode 4; and, the SG is required to be maintained at a level that ensures a significant inventory is available as a heat sink before the AFW pump is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1 (continued)

required to refill the SG. These factors ensure that a significant amount of time would be available to complete any valve realignments needed to refill a SG when in Mode 4.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code, Section XI (Ref. 2) (only required at 3 month intervals) satisfies this requirement.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test when SG pressure is < 600 psig.

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.3 (continued)

required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

This SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the required AFW train is operated as necessary to maintain SG water level.

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump is operated as necessary and the autostart function is not required. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

This SR is modified by two Notes. Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral allows the test to be performed at rated conditions. Note 2 states that the SR is not required in MODE 4. In MODE 4, the required pump is operated as necessary to maintain SG water level and the autostart function is not required. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 10.2.
2. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST)

BASES

BACKGROUND

The CST provides a safety grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves or the atmospheric dump valves. The AFW steam driven pump operates with a continuous recirculation to the CST. The motor driven AFW pumps have recirculation controllers that recirculate flow to the CST, as necessary, to maintain a minimum required AFW pump flow.

When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the steam bypass (High Pressure Steam Dump) valves. The condensed steam is returned to the CST by the condensate pump. This has the advantage of conserving condensate while minimizing releases to the environment.

Because the CST is a principal component in removing residual heat from the RCS, it is designed to withstand earthquakes and other natural phenomena. The CST is designed to Seismic Class I to ensure availability of the auxiliary feedwater supply. Auxiliary feedwater is also available from city water.

The condensate makeup system connects the 600,000 gallon capacity condensate storage tank to the main condenser. The condensate makeup system automatically supplies makeup water from the CST to the condenser if there is a low level in the condenser hotwell. Redundant, Category I, isolating valves will close the condenser makeup when the condensate storage tank level decreases to 360,000 gallons to reserve the required volume of condensate available to the auxiliary feedwater pumps sufficient to hold the plant at hot shutdown for 24 hours following a trip at full power.

(continued)

BASES

BACKGROUND (continued)

To ensure CST pressure is maintained within its design limits while limiting the amount of air in contact with the condensate, two Category I, 100% capacity breather valves are installed on the dome of the CST. CST venting is required for the CST to perform both its normal and emergency function. The venting function can be met by either of the CST breather valves or equivalent venting capacity.

A description of the CST is found in the FSAR, Section 10.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CST provides cooling water to remove decay heat and the minimum amount of water in the condensate storage tank is the amount needed to maintain the plant for 24 hours at hot shutdown following a trip from full power. When the condensate storage tank supply is exhausted, city water will be used.

The CST satisfies Criteria 2 and 3 of 10 CFR 50.36.

LCO

To satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat while in MODE 3 for 24 hours following a reactor trip from 102% RTP. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine. When the condensate storage tank supply is exhausted, city water will be used.

The CST level required is equivalent to a total volume of ≥ 360,000 gallons, which is based on holding the unit in MODE 3 for 24 hours. This basis is established in Reference 1. The CST total volume includes allowances for instrument accuracy and the unuseable volume in the CST.

(continued)

BASES

LCO (continued)

The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level. CST venting and pressure relief capability are required for the CST to perform both its normal and emergency function. The venting and pressure relief functions are satisfied by either of the CST breather valves or equivalent venting capacity.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST is required to be OPERABLE.

In MODE 5 or 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the CST is not OPERABLE, the OPERABILITY of the backup supply (city water) should be verified by administrative means immediately and once every 12 hours thereafter. OPERABILITY of the backup auxiliary feedwater supply means that LCO 3.7.7, City Water, is met and includes verification that the flow paths from city water to the AFW pumps are OPERABLE. The CST must be restored to OPERABLE status within 7 days. The immediate Completion Time for verification of the OPERABILITY of the backup water supply ensures that Condition B is entered immediately if both the CST and City Water are inoperable. The 7 day Completion Time for restoration of the CST is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the CST.

B.1 and B.2

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on the steam generator for heat removal, within 18 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.

If Condition B is entered when both the CST and City Water are not Operable, Conditions and Required Actions for LCO 3.7.5, Auxiliary Feedwater System, may be appropriate.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST contains the required volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CST level.

REFERENCES

1. FSAR, Section 10.2.

B 3.7 PLANT SYSTEMS

B 3.7.7 City Water (CW)

BASES

BACKGROUND

City Water is the backup to the Condensate Storage Tank (CST) as a water supply for the Auxiliary Feedwater System. The CST, the preferred source of water for the Steam Generators (SGs), is capable of holding up to 600,000 gallons and is sized to meet the normal operating and maintenance needs of the main steam system. LCO 3.7.6, Condensate Storage Tank, requires that a minimum water level is maintained in the CST that is sufficient to remove residual heat for 24 hours at hot shutdown conditions following a trip from full power. Only when the CST supply is exhausted or not available will city water be used to supply the Auxiliary Feedwater System.

When the main steam isolation valves are open, the preferred means of heat removal from the RCS is to discharge steam to the condenser via the non-safety grade turbine steam bypass valves (High Pressure Steam Dump) with water supplied from the CST to the SGs using the AFW System. The condensed steam is returned to the CST by the condensate pump. This configuration conserves condensate and minimizes releases to the environment. The CST is the preferred source of water for the SGs.

When the CST supply is exhausted, city water is used to supply the Auxiliary Feedwater System for decay heat removal and plant cooldown. CW, although aligned to the IP3 site, is normally isolated from the AFW pump suctions.

The City Water System includes the site city water header consisting of the 1.5 million gallon city water storage tank and the connection to the offsite water supply. A description of the CW system is found in FSAR, Section 10 (Ref. 1).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

CW can be used to provide cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in the FSAR; however, CW is used only when the CST is not available or depleted.

CW satisfies Criterion 3 of 10 CFR 50.36.

LCO

This LCO requires that the CW supply header is aligned to the AFW pump suction headers except for the onsite isolation valves, which are normally closed. The City Water Storage Tank is not required to contain a specific volume of water; however, the static head on CW supply from the CW storage tank is used to indicate that the CW supply header and CW System are aligned to the IP3 site and available for use.

The OPERABILITY of the CW is determined by maintaining the supply header pressure at or above the minimum required pressure and periodic verification that the required lineups can be established.

APPLICABILITY

City Water is required to be OPERABLE in MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal. In MODE 5 or 6, CW is not required because the SGs are not normally used to remove decay heat when in these MODES.

ACTIONS

A.1 and A.2

If the CW header pressure is not within limits or system lineups are not as required, CW cannot be assumed to be available if needed as a backup water source for the CST. With CW not available, OPERABILITY of the CST must be verified by administrative means immediately and once every 12 hours thereafter. Operability of the CST means that LCO 3.7.6, Condensate Storage Tank, is met. The immediate Completion Time for verification of the OPERABILITY of the CST ensures that Condition B is entered immediately if both the CST and City Water

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

are inoperable. This ensures that either the CST or CW is available for decay heat removal and to support a plant cooldown. CW must be restored to OPERABLE status within 7 days because CW is assumed to be available to supply the Auxiliary Feedwater System when the CST supply is exhausted. The 7 day Completion Time for restoration of CW is acceptable because the CST is OPERABLE and the low probability of an event requiring CW during the 7 day Completion Time.

B.1 and B.2

If CW cannot be restored to OPERABLE within the 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 4, without reliance on the steam generator for heat removal, within 18 hours. The 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.

If Condition B is entered when both the CST and City Water are not Operable, Conditions and Required Actions for LCO 3.7.5, Auxiliary Feedwater System, may be appropriate.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR verifies that CW header pressure is greater than 30 psig which provides a high degree of assurance that the offsite CW supply is available to the site and properly aligned. Operating experience has demonstrated that CW header pressure decays rapidly due to normal onsite consumption if the offsite supply is not properly aligned or pressurized. The 12 hour Frequency provides a high degree of assurance of rapid identification of the inoperability of CW.

(continued)

BASES

SURVEILLLANCE REQUIREMENTS (continued)

SR 3.7.7.2

This SR verifies that the valve that isolates Unit 3 from the site city water supply and the city water storage tank is open. This isolation valve, CT-49, in the IP1 Utility Tunnel, is also identified as valve FP-1227. This SR may be performed by Consolidated Edison personnel. The 31 day Frequency is acceptable because the valve is sealed open and because periodic verification provided by SR 3.7.7.1 provides a high degree of assurance that the valve is positioned properly.

SR 3.7.7.3

This SR verifies the ability to cycle each valve between CW and the AFW pump suction. These are the only valves required to operate to align CW to the AFW pump suction. The testing requirements and Frequency for this SR are in accordance with the Inservice Testing Program.

REFERENCES

1. FSAR, Chapter 10.

B 3.7 PLANT SYSTEMS

B 3.7.8 Component Cooling Water (CCW) System

BASES

BACKGROUND

The Component Cooling Water (CCW) System is a closed-loop cooling system that provides cooling water for systems and components important to safety that are located in the Primary Auxiliary Building, the Fuel Storage Building, and the Containment Building. The CCW System transfers its heat load to the Service Water System via CCW heat exchangers. The Service Water System is a once through cooling system that transfers its heat load to the ultimate heat sink, the Hudson River.

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components including the spent fuel storage pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System, and thus to the environment.

The CCW System consists of three pumps and two heat exchangers. These components are divided into two independent, full capacity cooling loops with each loop consisting of one pump and a heat exchanger. The third CCW pump can be aligned to replace the pump in either loop. Each of the three CCW pumps is powered from a separate safeguards power train.

The CCW loops are cross connected during normal and emergency operation; however, the cooling loads are divided between the two loops so that each loop is capable of supplying the necessary service to support continued containment sump and core recirculation following a LOCA while supplying normal loads. Operating CCW loops cross-connected allows use of either CCW heat exchanger to cool all normal and post accident heat loads. Any service water system pump can be used to support either or both CCW heat exchangers. Isolation valves allow each loop to be isolated and operated as an independent component cooling loop.

(continued)

BASES

BACKGROUND (continued)

This configuration facilitates detection of radioactivity entering the loop for leak detection or isolation of a piping or component failure during an event. A surge tank in each loop ensures that sufficient net positive suction head is available.

CCW pumps continue to operate following a safety injection signal without loss of offsite power (LOOP); however, CCW pumps are stripped and must be started as needed following any event that includes a LOOP. Note that the CCW pumps are not re-started during the injection phase; therefore, the water volume of the CCW system must act as a heat sink during the injection phase when the CCW pumps are not running. This is acceptable even though safety injection pump bearings are cooled by CCW because the cooling water is circulated by a booster pump directly connected to the injection pump motor shaft. During the injection phase, the Recirculation Pumps are cooled by the Auxiliary Component Cooling Water pumps, which are not governed by this LCO.

Additional information on the design and operation of the system, along with a list of the components served, is presented in the FSAR, Section 9.3 (Ref. 1). The principal safety related function of the CCW System is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. This may be during a normal or post accident cooldown and shutdown.

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is for one CCW loop to remove the post loss of coolant accident (LOCA) heat load from the containment sump during the recirculation phase. Any one of the three CCW pumps in conjunction with any one of the two CCW heat exchangers is sufficient to accommodate the normal and post accident heat load if the CCW system is operated as two cross connected loops. Either CCW pump in conjunction with either CCW heat exchanger or the third CCW pump in conjunction with either associated CCW heat exchanger is sufficient if the CCW loops are isolated.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Because the component cooling pumps do not run during the injection phase if the event is accompanied by a loss of offsite power, the water volume of the CCW system is used as a heat sink. This heat load causes a temperature rise of approximately 7°F per hour in the component cooling water with no credit taken for the water volume in the surge tank. With a minimum initial CCW temperature of 110°F at the start of the accident, 6 hours are available before the cooling water temperature reaches 150°F; 10 hours is available before reaching 180°F. Evaluations of the heat removal capability of the CCW system are contained in References 2 and 3.

The CCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

The CCW System also functions to cool the unit from RHR entry conditions (T_{avg} < 350°F), to MODE 5 (T_{avg} < 200°F), during normal and post accident operations. The time required to cool from 350°F to 200°F is a function of the CCW and RHR flow rate, service water flow rate and UHS temperature. One CCW loop is sufficient to remove decay heat during subsequent operations with T_{avg} < 200°F. This assumes a maximum service water temperature of 95°F occurring simultaneously with the maximum heat loads on the system.

The CCW System satisfies Criterion 3 of 10 CFR 50.36.

LCO

The CCW loops are independent of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a DBA, one CCW loop is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two loops of CCW must be OPERABLE. At least one CCW loop will operate during the recirculation phase assuming the worst case single active failure occurs coincident with a loss of offsite power.

(continued)

BASES

LCO (continued)

A CCW loop consists of any of the three CCW pumps in conjunction with a CCW heat exchanger.

A CCW loop is considered OPERABLE when:

a. The pump and associated surge tank are OPERABLE; and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

The isolation of CCW from components or systems may render those components or systems inoperable but does not affect the OPERABILITY of the CCW System.

Note that the auxiliary component cooling water pumps support the Containment Recirculation pumps only and are governed by LCO 3.5.2, ECCS - Operating.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be prepared to perform its post accident safety functions, primarily RCS heat removal, which is achieved by cooling the RHR heat exchanger.

In MODE 5 or 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," be entered if an inoperable CCW loop results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

If one CCW loop is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the

(continued)

BASES

ACTIONS

A.1 (continued)

remaining OPERABLE CCW loop is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE loop, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the CCW loop cannot be restored to OPERABLE status 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 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 3.7.8.1

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CCW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. Valves located inside containment are considered to be locked. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 (continued)

that those valves capable of being mispositioned are in the correct position. Valves that are throttled are verified by verification of required flow.

The 92 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.8.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.3 (continued)

for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 9.3.
2. FSAR, Section 6.2.
3. WCAP-12313, "Safety Evaluation for an Ultimate Heat Sink Temperature Increased to 95° at IP-3."

B 3.7 PLANT SYSTEMS

B 3.7.9 Service Water System (SWS)

BASES

BACKGROUND

The SWS provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the SWS also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

The SWS consists of two separate, 100% capacity, safety related, cooling water headers. Each header is supplied by three pumps and includes the piping up to and including the isolation valves on individual components cooled by the SW. Each of the 6 SWS pumps is equipped with rotary strainers and isolation valves.

SWS heat loads are designated as either essential or nonessential. The essential SWS heat loads are those which must be supplied with cooling water immediately in the event of a LOCA and/or loss of offsite power (LOOP). Examples of essential loads are the emergency diesel generators (EDGs), containment fan cooler units (FCUs) and control room air conditioning system (CRACS). The nonessential SWS heat loads are those which are required only following the switch over to the recirculation phase following a postulated LOCA. Examples of nonessential loads are the component cooling water (CCW) heat exchangers.

The FCUs are connected in parallel to the essential SWS header. Normal SWS flow to the FCUs is controlled by TCV-1103. Required ESFAS flow to all five FCUs is initiated when either of the redundant SWS to FCU ESFAS valves (TCV-1104 or TCV-1105) opens automatically in response to an ESFAS actuation signal.

The EDGs are connected in parallel to the essential SWS header. Required ESFAS flow to all three EDGs is initiated when either of the redundant SWS to EDG ESFAS valves (FCV-1176 or FCV-1176A) opens automatically in response to an ESFAS actuation which starts the EDGs.

(continued)

BASES

BACKGROUND (continued)

The CRACS are connected in parallel to the essential SWS header. Required ESFAS flow to both CRACS is provided continuously because the redundant SWS to CRACS valves (TCV-1310/1311 and TCV-1312/1313) have been modified to provide the required flow at all times.

Either of the two SWS headers can be aligned to supply the essential heat loads or the nonessential SWS heat loads. Both the essential and nonessential SWS HEADERS are operated to support normal plant operation and the plant response to accidents and transients. The SWS PUMPS associated with the SWS header designated as the essential header will start automatically. The SWS pumps associated with the SWS header designated as the nonessential header must be manually started when required following a LOCA.

The essential SWS heat loads can be cooled by any two of the three service water pumps on the essential header. The nonessential SWS heat loads can be cooled by any one of the three service water pumps on the nonessential header. To ensure adequate flow to the essential header, the essential and nonessential headers may be cross connected only as necessary while swapping the essential SWS header with the non essential SWS header.

Service water pump suctions are located below the mean sea level in the Hudson River, the ultimate heat sink. This configuration ensures adequate submergence of the SWS pump suctions.

Additional information about the design and operation of the SW, along with a list of the components served, is presented in the FSAR, Section 9.6, (Ref. 1). The principal safety related function of the SWS is the removal of decay heat from the reactor via the CCW System.

APPLICABLE SAFETY ANALYSES

The design basis of the SWS is as follows: post accident essential SWS heat loads can be cooled by any two of the three

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

service water pumps on the designated essential header; and, post accident nonessential SWS heat loads can be cooled by any one of the three service water pumps on the designated nonessential header. With the minimum number of pumps operating, the essential and nonessential headers of the SWS have the required capacity to remove core decay heat following a design basis LOCA as discussed in References 1, 2 and 3. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System by the ECCS pumps. The Service Water System was designed to fulfill required safety functions while sustaining: (a) the single failure of any active component used during the injection phase of a postulated LOCA with or without a LOOP, or (b) the single failure of any active or passive component used during the long-term recirculation phase with or without a LOOP.

The operating modes of the IP3 SWS are as follows: a) normal mode; b) post-LOCA injection mode; and, c) post-LOCA recirculation mode. The postulated failure conditions of the SWS must include consideration of the limiting case for each operating mode of the system which are as follows:

a. Loss of the 10 inch turbine building service water supply header during normal operation and a seismic event;
b. Loss of instrument air, during the post-LOCA injection phase concurrent with single active component failure.
c. Loss of a SWS pump on both the essential and nonessential headers (resulting from an EDG failure) during the post-LOCA recirculation phase.

The SW, in conjunction with the CCW System, also cools the unit from residual heat removal (RHR) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of CCW and RHR system flow, SWS flow and UHS temperature. The design assumes a maximum SWS temperature of 95°F occurring simultaneously with maximum heat loads on the system (Ref. 3).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SWS satisfies Criterion 3 of 10 CFR 50.36.

LCO

Three of the three SWS pumps associated with the SWS header designated as the essential header; and, two of the three SWS pumps associated with the SWS header designated as the nonessential header must be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, while sustaining: (a) the single failure of any active component used during the injection phase of a postulated LOCA with or without a LOOP, or (b) the single failure of any active or passive component used during the long-term recirculation phase with or without a LOOP.

An SWS header is considered OPERABLE during MODES 1, 2, 3, and 4 when:

a. The required number of pumps, consistent with the header's designation as the essential or nonessential header, are OPERABLE; and

b. The essential and nonessential headers are isolated from each other by at least one closed valve except as specified by the NOTE to the ACTIONS;

c. The associated piping, valves, instrumentation and controls required to perform the safety related function are OPERABLE.

The SWS to FCU valves (TCV-1104 or TCV-1105) and SWS to EDG valves (FCV-1176 or FCV-1176A) are OPERABLE when they open automatically in response to ESFAS actuation signal or are blocked open.

APPLICABILITY

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

(continued)

BASES

APPLICABILITY (continued)	In MODES 5 and 6, the OPERABILITY requirements of the SWS are determined by the systems it supports.

ACTIONS

The ACTIONS are modified by a Note that specifies that LCO 3.0.3 is not applicable for 8 hours while swapping the essential SWS header with the nonessential SWS header but only if LCO 3.7.9 will be met after the essential and non-essential header are swapped. This means that the essential and nonessential SWS headers may be cross-connected for up to 8 hours during transfer of the designated essential SWS header to the alternate SWS header. This is acceptable because the transfer is performed infrequently (i.e., approximately every 90 days) and the low probability of an event while the headers are cross connected.

A.1 and B.1

If one of the three required SWS pumps on the essential SWS header is inoperable (i.e., Condition A), three Operable pumps must be restored to the essential SWS header within 72 hours. Likewise, if one of the two required SWS pumps on nonessential SWS header is inoperable (i.e., Condition B), the header must be restored so that there are two Operable pumps for the nonessential SWS header within 72 hours. With one required SWS pump inoperable on either or both SWS headers, the remaining OPERABLE SWS pumps are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in an OPERABLE SWS pump could result in loss of SWS function. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE pump(s) in the same header, and the low probability of a DBA occurring during this time period.

C.1 and D.1

Required ESFAS flow to all three EDGs is initiated when either of the redundant SWS to EDG valves (FCV-1176 or FCV-1176A) opens automatically in response to an ESFAS actuation which starts the

(continued)

BASES

ACTIONS

C.1 and D.1 (continued)

EDGs. Similarly, required ESFAS flow to all five FCUs is initiated when either of the redundant SWS to FCU valves (TCV-1104 or TCV-1105) opens automatically in response to an ESFAS actuation signal. The SWS to FCU valves and SWS to EDG valves are OPERABLE when they open automatically in response to an ESFAS actuation signal or are blocked open.

If one of the redundant SWS to EDG valves is inoperable, a single failure of the redundant valve could result in the failure of all three EDGs shortly after the initiation of an event. If one of the redundant SWS to FCU valves is inoperable, a single failure of the redundant valve could result in the failure of all five FCUs. Therefore, a Completion Time of 12 hours is established to restore the required redundancy.

A 12 hour Completion Time is acceptable for the SWS to EDG valves because SWS to the EDGs is still available and the low probability of an event with a loss of offsite power during this period. A 12 hour Completion Time is acceptable for the SWS to FCU valves because SWS to the FCUs is still available, the availability of Containment Spray, and the low probability of an event during this period.

If both SWS to EDG valves or both SWS to FCU valves are inoperable, entry into LCO 3.0.3 is required.

E.1

If the SWS piping and valves are inoperable for reasons other than those listed in Conditions A, B, C or D, the SWS must be restored within 12 hours. This is necessary to ensure that repairs to affected portions of the SWS are completed in a timely manner. This Action also ensures no unnecessary transients (i.e. plant shutdown) are placed on the plant as a result of conditions in the SWS that may challenge OPERABILITY but do not result in a loss of function.

(continued)

BASES

ACTIONS

E.1 (continued)

A 12 hour Completion Time is acceptable for SWS piping and valves other than those listed in Conditions A, B, C, or D based on the low probability of an event during this period. Additionally, the 12 hour Completion Time allows the Operator to perform the evaluations and/or actions necessary for restoring the SWS OPERABILITY. This Action is in lieu of the potential for decreased safety as a result of diverting the Operator's attention to the actions associated with taking the unit to shutdown.

F.1 and F.2

If more than one required SWS pump in either the essential or the nonessential header is inoperable; or, if the flow path associated with either header is not capable of performing its safety function (e.g., both SWS to EDG valves or both SWS to FCU valves are inoperable), then the unit must be placed in a MODE in which the LCO does not apply.

Additionally, if an SWS header cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply.

To achieve the required 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.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR is modified by a Note indicating that the isolation of the SWS components or systems may render those components inoperable, but does not affect the OPERABILITY of the SW.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1 (continued)

Verifying the correct alignment for manual, power operated, and automatic valves in the SWS flow path provides assurance that the proper flow paths exist for SWS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 92 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.9.2

This SR verifies proper automatic operation of the SWS valves on an actual or simulated actuation signal. The SWS is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.3

This SR verifies proper automatic operation of the SWS pumps on an actual or simulated actuation signal. The SWS is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 9.6.
2. FSAR, Section 6.2.
3. WCAP-12313, "Safety Evaluation for an Ultimate Heat Sink Temperature Increase to 95°F at Indian Point 3."

B 3.7 PLANT SYSTEMS

B 3.7.10 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for processing and operating heat from safety related components during a transient or accident, as well as during normal operation. This is done by utilizing the Service Water System (SWS) and the Component Cooling Water (CCW) System.

The ultimate heat sink for IP3 is the Hudson River. The UHS and supporting structures are capable of providing sufficient cooling for thirty days and are sufficient to:

(a) Support simultaneous safe shutdown and cooldown of both operating nuclear units at the Indian Point site and maintain them in a safe condition, and

(b) In the event of an accident in one unit, support required response to that accident and permit simultaneous safe shutdown and cooldown of the remaining unit and maintain them in a safe shutdown condition.

The ultimate heat sink is capable of withstanding the effects of the most severe natural phenomena associated with the Indian Point site, other site related events and a single failure of man-made structural features.

The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

APPLICABLE SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation. Because IP3 uses the UHS

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

as the normal heat sink for condenser cooling via the Circulating Water System, unit operation at full power is its maximum heat load. Its maximum post accident heat load occurs shortly after a design basis loss of coolant accident (LOCA). Near this time, the unit switches from injection to recirculation and the containment cooling systems and containment recirculation system are required to remove the core decay heat.

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis, which include worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and worst case single active failure (e.g., single failure of a manmade structure). The UHS meets Regulatory Guide 1.27 (Ref.3), which requires a 30 day supply of cooling water in the UHS.

The UHS satisfies Criterion 3 of 10 CFR 50.36.

LCO

The UHS is required to be OPERABLE and is considered OPERABLE if it contains water at or below the maximum temperature that would allow the SWS to operate for at least 30 days following the design basis LOCA without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the UHS temperature must not exceed 95°F.

APPLICABILITY

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

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

(continued)

BASES

ACTIONS

A.1 and A.2

If UHS temperature > 95°F, or is inoperable for reasons other than high temperature, 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 7 hours and in MODE 5 within 37 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

This SR verifies that the SWS is available to cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the average water temperature of the UHS is ≤ 95°F. Requirements for UHS monitoring instrumentation are governed by the Technical Requirements Manual (Ref. 4).

REFERENCES

1. FSAR, Section 9.6.
2. WCAP-12313, "Safety Evaluation For An Ultimate Heat Sink Temperature Increase To 95°F At Indian Point Unit 3"
3. Regulatory Guide 1.27.
4. IP3 Technical Requirements Manual.

B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Ventilation System (CRVS)

BASES

BACKGROUND

The CRVS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity, chemicals, or toxic gas.

The Control Room Ventilation System consists of the following equipment: a single filter unit consisting of two roughing filters, two high efficiency particulate air (HEPA) filters; two activated charcoal adsorbers for removal of gaseous activity (principally iodines); two 100% capacity filter booster fans; and, a single duct system including dampers, controls and associated accessories to provide for three different air flow configurations. The air-conditioning units associated with the CRVS are governed by LCO 3.7.12, "Control Room Air Conditioning System (CRACS)."

The CRVS is divided into two trains with each train consisting of a filter booster fan and the associated inlet damper and the following components which are common to both trains: the control room filter unit, damper A (filter unit bypass for outside air makeup to the Control Room), damper B (filter unit inlet for outside air makeup to the Control Room), damper C (filter unit inlet for reticulated air), and the toilet and locker room exhaust fan. The two filter booster fans (F 31 and F 32) are powered from safeguards power trains 5A (EDG 33) and 6A (EDG 32), respectively. The automatic dampers that are common to both trains are positioned in the fail-safe position (open or closed) by either of the redundant actuation channels.

The CRVS is an emergency system, parts of which operate during normal unit operations.

The three different CRVS air flow configurations are as follows:

(continued)

BASES

BACKGROUND (continued)

a) Normal operation consists of approximately 85% (8500 cfm) unfiltered recirculated flow driven by the air-conditioning fans and approximately 15% (1500 cfm) unfiltered outside air makeup;

b) Incident mode with outside air makeup (i.e. 10% incident mode) consists of approximately 87% (9250 cfm) unfiltered recirculated flow driven by the two safety related air-conditioning fans, at least 10% (> 1000 cfm) filtered recirculated flow driven by either one of the two filter booster fans and approximately 2.5% to 4.0% (250 to 400 cfm) filtered outside air makeup;

c) Incident mode with no outside air makeup (i.e. 100% incident mode) consists of 85% (9100 cfm) unfiltered recirculated flow driven by the two safety related air-conditioning fans, approximately 15% filtered recirculated flow driven by either one of the two filter booster fans and no outside air makeup.

Note that the required recirculation rates are demonstrated with surveillance tests conducted with the air conditioning system (CRACS) operating. An inoperable CRACS fan will affect the flow balance of the CRVS due to interconnected ductwork. Therefore, if the fan associated with one of the air-conditioning units governed by LCO 3.7.12 is inoperable, Conditions in both LCO 3.7.11, Control Room Ventilation System, and LCO 3.7.12, Control Room Air Conditioning System (CRACS), will apply.

Incident mode with outside air makeup is the preferred method of operation during any radiological event because it provides outside air for pressurization of the Control Room. Calculations indicate that very low volumes of outside air makeup will maintain the Control Room at a slight positive pressure. Nevertheless, due to the difficulty of adjusting and maintaining the flow dampers to provide a low flow, it was determined that the damper should be adjusted to provide a flow of approximately 250 cfm (2.5% outside air makeup). However, a higher volume of outside air makeup to

(continued)

BASES

BACKGROUND (continued)

the Control Room increase the thyroid dose to the operators during an accident. Therefore, the Control Room dose assessment assumes a filtered outside air makeup of approximately 400 cfm (4.0% outside air makeup).

On a Safety Injection signal or high radiation in the Control Room (Radiation Monitor R-1), the CRVS will actuate to the incident mode with outside air makeup (i.e. 10% incident mode). This will cause one of the two filters booster fans to start, the locker room exhaust fan to stop, and CRVS dampers to open or close as necessary to filter all incoming outside air and direct approximately 10% of the recirculated air through the filter unit. In the event that the first booster fan fails to start, the second booster fan will start after a predetermined time delay.

If for any reason it is required or desired to operate with 100% recirculated air (e.g., toxic gas condition is identified), the CRVS can be placed in the incident mode with no outside air makeup (i.e. 100% incident mode) by remote manually operated switches. The Firestat detectors will also initiate 100% incident mode in the CRVS.

The control room is continuously monitored by radiation and toxic gas detectors. On a Safety Injection signal or high radiation in the Control Room (Radiation Monitor R-1), will cause actuation of the emergency radiation state of the CRVS (i.e., incident mode with outside air makeup (i.e. 10% incident mode)).

The CRVS does not actuate automatically in response to toxic gases. Separate chlorine, ammonia and oxygen probes are provided to detect the presence of these gases in the outside air intake. Additionally, monitors in the Control Room will detect low oxygen levels and high levels of chlorine and ammonia. The CRVS may be placed in the incident mode with no outside air makeup (i.e. 100% incident mode) to respond to these conditions. Instrumentation for toxic gas monitoring is governed by the IP3 Technical Requirements Manual (TRM) (Ref. 4). Generally, the manually initiated actions of the toxic gas isolation state are more restrictive, and will override the actions of the emergency radiation state.

(continued)

BASES

BACKGROUND (continued)

A single train will create a slight positive pressure in the control room. The CRVS operation in maintaining the control room habitable is discussed in the FSAR, Section 9.9 (Ref. 1).

The CRVS is designed in accordance with Seismic Category I requirements.

The CRVS is designed to maintain the control room environment for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem whole body dose or 30 rem to the thyroid.

APPLICABLE SAFETY ANALYSES

The CRVS active components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the control building envelope ensures an adequate supply of filtered air to all areas requiring access. The CRVS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis accident (i.e., DBA LOCA) fission product release presented in the FSAR, Chapter 14 (Ref. 2).

Radiation monitor R-1 is not required for the Operability of the Control Room Ventilation System because control room isolation is initiated by the safety injection signal in MODES 1, 2, 3, 4, and control room isolation is not required for maintaining radiation exposure within General Design Criteria 19 limits following a fuel handling accident or gas-decay-tank rupture.

The worst case active failure of a component of the CRVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. However, the original CRVS design was not required to meet single failure criteria and, although upgraded from the original design, CRVS does not satisfy all requirements in IEEE-279 for single failure tolerance.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Each of the automatic dampers that are common to both trains is positioned in the fail-safe position (open or closed) by either of the redundant actuation channels.

The CRVS satisfies Criterion 3 of 10 CFR 50.36.

LCO

Two CRVS trains are required to be OPERABLE to ensure that at least one is available. Total system failure could result in exceeding a dose of 5 rem whole body or 30 rem to the thyroid of the control room operator in the event of a large radioactive release.

The CRVS is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in both trains. A CRVS train is OPERABLE when the associated:

a. Filter booster fan and an air-conditioning unit fan powered from the same safeguards power train are OPERABLE;

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

c. Ductwork, valves, and dampers are OPERABLE or in the incident mode, and air circulation can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

Instrumentation for toxic gas monitoring is governed by the IP3 Technical Requirements Manual (TRM) (Ref. 4) and is not included in the LCO.

Note that the required recirculation rates are demonstrated with surveillance tests conducted with the air conditioning system (CRACS) operating. An inoperable CRACS fan will affect the flow

(continued)

BASES

LCO (continued)

balance of the CRVS due to interconnected ductwork. Therefore, if the fan associated with one of the air-conditioning units governed by LCO 3.7.12 is inoperable, Conditions in both LCO 3.7.11, Control Room Ventilation System, and LCO 3.7.12, Control Room Air Conditioning System (CRACS), will apply.

APPLICABILITY

In MODES 1, 2, 3, 4 CRVS must be OPERABLE to limit operator exposure during and following a DBA.

The CRVS is not required in MODE 5 or 6, or during movement of irradiated fuel assemblies and core alterations because analysis indicates that isolation of the control room is not required for maintaining radiation exposure within acceptable limits following a fuel handling accident or gas decay tank rupture.

Administrative controls address the role of the CRVS in maintaining control room habitability following an event at Indian Point Unit 2.

ACTIONS

A.1

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

B.1

When neither CRVS train is Operable, action must be taken to restore at least one train to OPERABLE status within 72 hours. The 72 hour Completion Time is acceptable because of the low probability of a DBA occurring during this time period.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If Required Actions A.1 or B.1 are not met 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.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Note that a CRVS train includes both the filter booster fan and an air-conditioning unit fan powered from the same safeguards power train. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy availability.

SR 3.7.11.2

This SR verifies that the required CRVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CRVS filter tests are in accordance with the sections of Regulatory Guide 1.52 (Ref. 3) identified in the VFTP. 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.11.3

This SR verifies that each CRVS train starts and operates on an actual or simulated actuation signal. The Frequency of 24 months is based on operating experience which has demonstrated this Frequency provides a high degree of assurance that the booster fans will operate and dampers actuate to the correct position when required.

SR 3.7.11.4

This SR verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the CRVS. During the operation in the incident mode with outside air makeup (i.e. 10% incident mode), the CRVS is designed to maintain the control room at a slight positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The CRVS is designed to maintain this positive pressure with very low volumes of outside air makeup. Due to the difficulty of adjusting and maintaining the flow dampers to provide a low flow, it was determined that the damper should be adjusted to provide a flow of approximately 250 cfm (2.5% outside air makeup). Note that the higher the volume of outside air makeup to the Control Room, the higher the thyroid dose to the operators during an accident. The acceptance criteria of 400 cfm (4.0% outside air makeup) is the volume used in the Control Room dose assessment.

The SR Frequency of 24 months on a staggered test basis is acceptable because operating experience has demonstrated that the control room boundary is not normally disturbed. Staggered testing is acceptable because the SR is primarily a verification of Control Room integrity because fan operation is tested elsewhere.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 9.9.
2. FSAR, Chapter 14.
3. Regulatory Guide 1.52, Rev. 2.
4. IP3 Technical Requirements Manual.

B 3.7 PLANT SYSTEMS

B 3.7.12 Control Room Air Conditioning System (CRACS)

BASES

BACKGROUND

The CRACS provides temperature control for the control room following isolation of the control room.

The CRACS consists of two trains that provide cooling of recirculated control room air. Each train consists of, cooling coils, instrumentation, and controls to provide for control room temperature control. The CRACS (CRACS 31 and CRACS 32) are powered from safeguards power trains 5A (EDG 33) and 6A (EDG 32), respectively. The CRACS units are supplied with cooling water from the essential service water header and each unit is capable of performing its design function during an accident with a service water inlet temperature ≤ 95°F.

The CRACS is an emergency system, parts of which may also operate during normal unit operations. Each CRACS unit is sized to provide 60% of the cooling capacity required during normal operation and 100% of the cooling capacity required during an accident. The CRACS operation in maintaining the control room temperature is discussed in the FSAR, Section 9.9 (Ref. 1).

During normal operation, five supplemental air-conditioning units in the Control Room are available to supplement the cooling capacity of the CRACS. These units also provide Control Room heating. These five supplemental air-conditioning units are not assumed to be available during a blackout or design basis accident and, therefore, are not governed by Technical Specifications.

APPLICABLE SAFETY ANALYSES

The design basis of the CRACS is to maintain the control room temperature for 30 days of continuous occupancy.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The CRACS components are arranged in redundant, safety related trains. The CRACS is designed so that the functional capacity of the Control Room is maintained at all times, including a Design Basis Accident. Functional capacity of the Control Room means that the ambient temperature for safety equipment located in this room will not exceed 108.2°F. Control Room safety equipment is specified to a temperature of 120°F and the 108.2°F limit for Control room temperature is sufficient to account for the temperature rise in the enclosed cabinets. Functional capacity of the Control Room can be maintained by one train of CRACS being cooled by the essential service water system assuming the ultimate heat sink temperature is ≤ 95°F. Analysis indicates that under worst case conditions, the Control Room temperature could rise to approximately 106°F following the loss of one CRACS train assuming all lights, except emergency lights, are turned off (Ref.1). Detectors and controls are provided for control room temperature control. The CRACS is designed in accordance with Seismic Category I requirements. The CRACS is capable of removing sensible and latent heat loads from the control room, which include consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.

A failure of a component of the CRACS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. However, the original CRACS design was not required to meet single failure criteria and, although upgraded from the original design, CRACS does not satisfy all requirements in IEEE-279 for single failure tolerance.

The CRACS satisfies Criterion 3 of 10 CFR 50.36.

LCO

Two trains of the CRACS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disabling the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

(continued)

BASES

LCO (continued)

The CRACS is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both trains. These components include the cooling coils and common temperature control instrumentation. In addition, the CRACS must be operable to the extent that air circulation can be maintained.

Note that the required recirculation rates are demonstrated with surveillance tests conducted with the air conditioning system (CRACS) operating. An inoperable CRACS fan will affect the flow balance of the CRVS due to interconnected ductwork. Therefore, if the fan associated with one of the air-conditioning units governed by LCO 3.7.12 is inoperable, Conditions in both LCO 3.7.11, Control Room Ventilation System, and LCO 3.7.12, Control Room Air conditioning System (CRACS), will apply.

APPLICABILITY

In MODES 1, 2, 3 and 4, the CRACS must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements following isolation of the control room.

The CRACS is not required in MODE 5 or 6, or during movement of irradiated fuel assemblies and core alterations because analysis indicates that isolation of the control room is not required for maintaining radiation exposure within acceptable limits following a fuel handling accident or gas decay tank rupture.

ACTIONS

A.1

With one CRACS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CRACS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE CRACS train could result in loss of CRACS function. The 30 day Completion Time is based on the low probability of an event requiring control room isolation, the consideration that the remaining train can provide the required protection, and that alternate nonsafety related cooling means are typically available.

(continued)

BASES

ACTIONS (continued)

B.1

When neither CRACS train is Operable, action must be taken to restore at least one train to OPERABLE status within 72 hours. The 72 hour Completion Time is acceptable because of the low probability of a DBA occurring during this time period and because alternate nonsafety cooling means are typically available.

C.1 and C.2

If Required Actions A.1 or B.1 are not met within the required Completion Time, the unit must be placed in a MODE that minimizes the 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.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load required to maintain functional capacity of the Control Room at all times (Ref. 1). This SR consists of a combination of testing and calculations. The 24 month Frequency is appropriate since significant degradation of the CRACS is slow and is not expected over this time period.

REFERENCES

1. FSAR, Section 9.9.

B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Storage Building Emergency Ventilation System (FSBEVS)

BASES

BACKGROUND

The FSBEVS filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident. The FSBEVS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the fuel storage building.

The Fuel Storage Building (FSB) ventilation system maintains environmental conditions in the building enclosing the spent fuel pit and consists of the following:

Two FSB air tempering units with associated ventilation supply fans and ventilation supply isolation dampers;

One FSB exhaust fan and associated outlet damper;

One FSB exhaust filtration unit consisting of roughing, HEPA, and charcoal filters which includes the pneumatically operated inlet and outlet dampers for the carbon filter and manually operated dampers that allow the carbon filter to be bypassed;

Inflatable seals on man doors and truck door,

Area Radiation Monitor (R-5) consisting of an extended range area monitor used to measure the area radiation fields of the Fuel Storage Building; and,

Ductwork, dampers, and instrumentation needed to support system operation,

<u>During normal operation</u>, the FSB air tempering units and associated ventilation supply fans and the FSB exhaust fan operate, as necessary, to ventilate and, if necessary, heat the FSB. One or both FSB air tempering units are used to supply outside air to the south end of the FSB and the FSB exhaust fan

(continued)

BASES

BACKGROUND (continued)

is used to exhaust air from the north end of the FSB through the roughing filters and HEPA filters and is released to the environment via the plant vent. FSB air flow is directed from radiologically clean to less clean areas to prevent the spread of contamination. Additionally, the FSBEVS is designed so that the exhaust fan capacity is greater than the supply fan(s) capacity so that the FSB is normally maintained at a slight negative pressure. This ensures that ventilation air leaving the FSB passes through the filters and HEPA in the exhaust filtration unit and is released to the environment via the plant vent. When not handing irradiated fuel in the FSB, the carbon filter in the exhaust filtration unit is normally bypassed to extend the life of the charcoal. In this configuration, the manually operated charcoal filter bypass dampers are left open and the automatically operated charcoal filter face dampers (inlet and outlet dampers) are closed.

<u>During irradiated fuel handling activities in the FSB,</u> the FSBEVS is operated as described above except that the manually operated charcoal filter bypass dampers are closed and the charcoal filter face dampers (inlet and outlet dampers) are opened. In this configuration, the FSB is still maintained at a slight negative pressure but all FSB ventilation exhaust is directed through the roughing filters, HEPA filters, and charcoal filters and is released to the environment via the plant vent.

<u>Following an Area Radiation Monitor (R-5) signal or manual actuation to the emergency mode of operation</u>, the ventilation supply fans stop automatically and the associated ventilation supply dampers close automatically. The charcoal filter face dampers (inlet and outlet dampers) open automatically, if not already open. Additionally, the rolling truck door closes, if open, and the inflatable seals on the man doors and truck door are actuated. The FSB exhaust fan continues to operate. With the FSB ventilation supply stopped and the FSB boundary secured, the FSB exhaust fan is capable of maintaining the FSB at a pressure $\leq$ -0.5 inches water gauge with respect to atmospheric pressure with the exhaust flow rate $\leq$ 20,000 cfm. Ventilation dampers required to establish the boundary or flow path (e.g., air tempering unit ventilation supply inlet dampers) will fail·

(continued)

BASES

BACKGROUND (continued)

safe into the required emergency mode position. Note that the inflatable seals on man doors and truck door are not required for maintaining the FSB at these required post accident conditions.

A push button switch adjacent to the 95' elevation door leading to the Fan House allows the Fuel Storage Building Exhaust Fan to be momentarily shut down and air removed from the man door seal to allow the door to be opened for FSB ingress or egress when in the emergency mode of operation. The fan will automatically restart and the door is resealed after a preset time has elapsed (approximately 30 seconds).

The FSBEVS is discussed in the FSAR, Sections 9.5, and 14.2 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The FSBEVCS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis for a fuel handling accident assumes that the FSB exhaust fan can maintain the FSB at a slight negative pressure (i.e., $\leq$ -0.125 inches water gauge) with respect to atmospheric pressure with the exhaust flow rate $\leq$ 20,000 cfm. Under these conditions, all FSB ventilation exhaust is assumed to be directed through the roughing filters, HEPA filters, and charcoal filters and is released to the environment via the plant vent. This ensures that offsite post accident dose rates are within required limits. Although this LCO requires the OPERABILITY of the FSBEVS whenever irradiated fuel assemblies are being moved within the FSB, analysis indicates that offsite post accident dose rates will be within required limits without the operation of the FSBEVS if the irradiated fuel has had a continuous 45 day decay period. This analysis is described in Reference 2.

The FSBEVS satisfies Criterion 3 of 10 CFR 50.36.

(continued)

BASES (continued)

LCO

This LCO requires that the Fuel Storage Building Emergency Ventilation System is OPERABLE and the FSB boundary is intact. This ensures that the required negative pressure is maintained in the FSB and FSB ventilation exhaust is directed through the roughing filters, HEPA filters, and charcoal filters and is released to the environment via the plant vent. Failure of the FSBEVS or the FSB boundary could result in the atmospheric release from the fuel storage building exceeding the 10 CFR 100 (Ref. 3) limits in the event of a fuel handling accident.

The FSBEVS is considered OPERABLE when the individual components necessary to control exposure in the fuel storage building are OPERABLE. FSBEVS is considered OPERABLE when its associated:

a. Exhaust fan is OPERABLE;

b. Roughing filter, HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function;

c. Ductwork and dampers are OPERABLE as needed to ensure air circulation can be maintained through the filter;

d. Ventilation supply fan trip function and ventilation supply isolation dampers closure function are OPERABLE or secured in incident position; and

e. FSBEVS charcoal filter bypass dampers are closed and leak tested.

The inflatable seals on man doors and truck door are not required for maintaining the FSB at these required post accident conditions. Additionally, the FSBEVS is not rendered inoperable when the FSBEVS exhaust fan is momentarily shut down and air removed from the door seal to allow the door to be opened for FSB ingress or egress when in the emergency mode of operation.

Requirements for the OPERABILITY of the Area Radiation Monitor (R-5) and associated instrumentation that initiates the FSBEVS are addressed in LCO 3.3.8, "Fuel Storage Building Emergency Ventilation System Actuation Instrumentation."

(continued)

BASES

LCO (continued)

Requirements for leak testing the FSBEVS charcoal filter bypass dampers following closure are governed by the IP3 FSAR.

APPLICABILITY

During movement of irradiated fuel in the fuel storage building, the FSBEVS is required to be OPERABLE to mitigate the consequences of a fuel handling accident.

ACTIONS

A.1

When the FSBEVS is inoperable during movement of irradiated fuel assemblies in the fuel storage building, action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the fuel storage building. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR requires periodic verification that the FSBEVS charcoal filter bypass dampers are installed and leak tested. This SR is performed by a visual verification that the bypass dampers are installed and an administrative verification that required leak testing was performed following the last installation of the dampers. Requirements for leak testing the FSBEVS charcoal filter bypass dampers following closure are governed by the IP3 FSAR.

This SR is performed prior to movement of irradiated fuel assemblies in the fuel storage building, and once per 92 days thereafter. The 92 day Frequency is appropriate because the bypass dampers are operated under administrative controls which provides a high degree of assurance that the dampers will remain in the required position. This Frequency has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.2

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing the FSBEVS once every 31 days provides an adequate check on this system. Systems are operated for ≥ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the known reliability of the equipment.

SR 3.7.13.3

This SR verifies that the required FSBEVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The FSBEVS filter tests are in accordance with the applicable portions of Regulatory Guide 1.52 (Ref. 4) as specified in the VFTP. The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.13.4

This SR verifies that the FSBEVS starts and operates on an actual or simulated actuation signal. The 92 day Frequency ensures that the SR is performed within a short time prior to a potential need for the FSBEVS and allows the SR to be performed only once prior to or during a refueling outage. This SR Frequency is based on the demonstrated reliability of the system.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.5

This SR verifies the integrity of the fuel storage 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 FSBEVS. During the normal mode of operation, the FSBEVS is designed to maintain a slight negative pressure in the fuel storage building, to prevent unfiltered LEAKAGE. This test verifies that the FSB exhaust fan can maintain the FSB at a slight negative pressure (i.e., ≤ -0.125 inches water gauge) with respect to atmospheric pressure with the exhaust flow rate ≤ 20,000 cfm during a fuel handling accident. The Frequency of 24 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 5).

REFERENCES

1. FSAR, Section 9.5.
2. FSAR, Section 14.2.
3. 10 CFR 100.
4. Regulatory Guide 1.52 (Rev. 2).
5. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Pit Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel pit meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the spent fuel pit design and the Spent Fuel Cooling and Cleanup System is given in the FSAR, Section 9.5 (Ref. 1). The assumptions of the fuel handling accident are given in the FSAR, Section 14.2 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The minimum water level in the spent fuel pit meets the assumptions of the fuel handling accident described in FSAR, Section 14.2 (Ref. 2). The resultant 2 hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 100 (Ref. 3) limits.

According to Reference 2, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 2 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks.

The Spent Fuel Pit water level satisfies Criteria 2 and 3 of 10 CFR 50.36.

(continued)

BASES (continued)

LCO	The spent fuel pit water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 2). As such, it is the minimum required for fuel storage and movement within the spent fuel pit.
APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the spent fuel pit, since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel pit water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel pit is immediately suspended to a safe position. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies sufficient spent fuel pit water is available in the event of a fuel handling accident. The water level in the spent fuel pit must be checked periodically. The 7 day Frequency is appropriate because the volume in the spent fuel pit is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

During refueling operations, the level in the spent fuel pit is normally in equilibrium with the refueling canal and reactor cavity, and the level in the refueling reactor cavity is checked daily in accordance with SR 3.9.6.1.

REFERENCES

1. FSAR, Section 9.5.
2. FSAR, Section 14.2.
3. 10 CFR 100.11.

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Pit Boron Concentration

BASES

BACKGROUND

In the Maximum Density Rack (MDR) design, the spent fuel storage pool is divided into two separate and distinct regions. The layout of the IP3 MDR is shown in Figure B 3.7.16-1. As shown in Figure B 3.7.16-1, Region 1 (Columns SS-ZZ, Rows 35-64) includes 240 storage positions and Region 2 (Columns A-RR, Rows 1-34) includes 1105 storage positions. Region 1 is analyzed for storage of high-enrichment and low-burnup fuel. Region 2 is analyzed for storage of fuel with either higher burnup or lower enrichment. Each region has been separately analyzed for close packed storage when all cells in that region contain fuel of the highest reactivity stored in accordance with LCO 3.7.16, Spent Fuel Assembly Storage. This analysis is the basis for the restrictions on fuel storage locations established by LCO 3.7.16.

Limits, based on a combination of initial enrichment and burnup, are used to determine if a fuel assembly must be stored in region 1 or if the fuel assembly may be stored in either region 1 or region 2. Fuel with the highest initial enrichments are subject to additional restrictions even when stored in region 1. Fuel assemblies with an initial enrichment > 5.0 wt% U-235 cannot be stored in the spent fuel pit in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.

The water in the spent fuel pit normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded when fuel storage locations, enrichment and burnup are in conformance with analysis assumptions as specified in LCO 3.7.16. The double contingency

(continued)

BASES

BACKGROUND (continued)

principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, because only a single accident need be considered at one time. For example, the accident scenarios include movement of fuel from Region 1 to Region 2, or accidental misloading of a fuel assembly in Region 1. This event could increase the potential for criticality of the spent fuel pit. To mitigate these postulated criticality related accidents, boron concentration is verified by SR 3.7.15.1 to be within the limits specified in this LCO prior to movement of fuel assemblies in the spent fuel pit. Safe operation of the MDR with no movement of assemblies is achieved by controlling the location of each assembly in accordance with LCO 3.7.16, "Spent Fuel Assembly Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.15.1.

APPLICABLE SAFETY ANALYSES

Most accident conditions do not result in an increase in the reactivity of either of the two regions. Examples of these accident conditions are the loss of cooling (reactivity increase with decreasing water density) and the dropping of a fuel assembly on the top of the rack. However, accidents can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accident occurrences, the presence of soluble boron in the storage pool prevents criticality in both regions. The postulated accidents are basically of two types. A fuel assembly could be incorrectly transferred from Region 1 to Region 2 (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). The second type of postulated accidents is associated with a fuel assembly which is dropped adjacent to the fully loaded storage rack. This could have a small positive reactivity effect in the Region. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. The accident analyses is described in References 2 and 3.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The concentration of dissolved boron in the spent fuel pit satisfies Criterion 2 of 10 CFR 50.36.

LCO

The spent fuel pit boron concentration is required to be ≥ 1000 ppm. The specified concentration of dissolved boron in the spent fuel pit preserves the assumptions used in the analyses of the potential critical accident scenarios as described in Reference 3. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pit until a spent fuel pit verification confirms that there are no mis-loaded fuel assemblies. With no mis-loaded fuel assemblies and unborated water, the spent fuel pit design is sufficient to maintain the core at $k_{eff} \leq 0.95$.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pit, until a complete spent fuel pit verification has been performed following the last movement of fuel assemblies in the spent fuel pit. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

A.1, A.2.1 and A.2.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the spent fuel pit is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending

(continued)

BASES

ACTIONS

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

movement of fuel assemblies. Alternatively, beginning a verification of the Spent Fuel Pit fuel locations, to ensure proper locations of the fuel, can be performed. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the spent fuel pit is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 31 day Frequency is appropriate because no major replenishment of spent fuel pit water is expected to take place over such a short period of time. This SR is not required to be met or performed if a spent fuel pit verification for conformance with LCO 3.7.16, Figures 3.7.16-1 and B 3.7.16-1, has been performed on all fuel assemblies since the last verification following the last movement of fuel assemblies in the spent fuel pit.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

(continued)

BASES

REFERENCES (continued)

2. SER related to Amendment 173 to Facility Operating License No. DPR-64, Indian Point Nuclear Generating Unit No. 3, April 15, 1997.

3. Criticality Analysis of the Indian Point 3 Fresh and Spent Fuel Racks, Westinghouse Commercial Nuclear Fuel Division, October, 1996.

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Assembly Storage

BASES

BACKGROUND

In the Maximum Density Rack (MDR) design, the spent fuel pit (SFP) is divided into two separate and distinct regions. The layout of the IP3 MDR is shown in Figure B 3.7.16-1, IP3 Maximum Density Spent Fuel Pit Racks, Regions and Indexing. As shown in Figure B 3.7.16-1, Region 1 (i.e., Columns SS-ZZ, Rows 35-64) includes 240 storage positions and Region 2 (i.e., Columns A-RR, Rows 1-34) includes 1105 storage positions. Region 1 is analyzed for storage of high-enrichment and low-burnup fuel. Region 2 is analyzed for storage of fuel with either higher burnup or lower enrichment. Each region has been separately analyzed for close packed storage when all cells in that region contain fuel of the highest reactivity that is allowed by this LCO. This analysis is the basis for the restrictions on fuel storage locations established by this LCO.

Prior to storage in the spent fuel pit, fuel assemblies are classified as to the level of reactivity based on the initial enrichment and burnup. This classification is made using Figure 3.7.16-1, "Fuel Assembly Classification for Storage in the Spent Fuel Pit". This classification is used to determine in which region a particular fuel assembly may be stored and if additional restrictions must be applied to the assemblies in adjacent locations. Figure 3.7.16-1, "Fuel Assembly Classification for Storage in the Spent Fuel Pit", is used to classify each assembly into one of the following categories based on initial U-235 enrichment and burnup:

Type 2 assemblies are the least reactive assemblies and include any assembly for which the combination of initial enrichment and burnup places the assembly in the domain labeled Type 2 in Figure 3.7.16-1. Type 2 assemblies may be stored in any location in Region 1 or Region 2 of Figure B 3.7.16-1.

Type 1A assemblies are more reactive than Type 2 assemblies and include any assembly for which the combination of initial enrichment and burnup places the assembly in the domain labeled

(continued)

BASES

BACKGROUND (continued)

Type 1A in Figure 3.7.16-1. Type 1A assemblies must be stored in Region 1 of Figure B 3.7.16-1 but may be stored in any location in Region 1.

Type 1B assemblies are more reactive than Type 1A assemblies and include any assembly with an initial enrichment > 4.2 but ≤ 4.6 wt% U-235 with a burnup that places the assembly in the domain labeled Type 1B in Figure 3.7.16-1. Type 1B assemblies must be stored in Region 1 of Figure B 3.7.16-1 but may be stored in any location in Region 1 except in locations that are face-adjacent to a Type 1C assembly.

Type 1C assemblies are the most reactive bundles permitted in accordance with Specification 4.3, Fuel Storage. Type 1C assemblies include any assembly with an initial enrichment > 4.6 but ≤ 5.0 wt% U-235 with a burnup that places the assembly in the domain labeled Type 1C on Figure 3.7.16-1. Type 1C assemblies must be stored in Region 1 of Figure B 3.7.16-1. Type 1C assemblies cannot be stored in Row 64 or in Column ZZ. Additionally, Type 1C assemblies must be stored in a location where all face-adjacent locations are as follows:

a) occupied by Type 2 or Type 1A assemblies;

b) occupied non-fuel components; or, c) empty.

Fuel assemblies with an initial enrichment > 5.0 wt% U-235 are not shown on Figure 3.7.16-1 and cannot be stored in the spent fuel pit in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.

The water in the spent fuel pit normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded and fuel storage locations, enrichment and burnup are in conformance with analysis assumptions and this LCO. The double contingency principle

(continued)

BASES

BACKGROUND (continued)

discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions because only a single accident need be considered at one time. For example, the accident scenarios include movement of a type 1C fuel assembly from Region 1 to Region 2, or accidental misloading of a fuel assembly in Region 1. These events could increase the potential for criticality in the Spent Fuel Pit. To mitigate these postulated criticality related accidents, boron concentration is verified to be within the limits specified in LCO 3.7.15, Spent Fuel Pit Boron Concentration, prior to movement of any fuel assembly. Safe operation of the SFP with no movement of assemblies is achieved by controlling the location of each assembly in accordance with the accompanying LCO. However, prior to movement of an assembly, it is necessary to perform SR 3.7.15.1 (i.e., verification that the spent fuel pit boron concentration is within limit).

APPLICABLE SAFETY ANALYSES

The restrictions on the placement of fuel assemblies within the spent fuel pit are based on initial enrichment and burnup which is indicative of fuel assembly reactivity. Storage locations are then restricted to ensure the k_{eff} of the spent fuel pit will always remain < 0.95, assuming the pool to be flooded with unborated water. Fuel assemblies not meeting the criteria of Figure 3.7.16-1 may not be stored in accordance with Specification 4.3.1.1 in Section 4.3.

The hypothetical accidents can only take place during or as a result of the movement of an assembly (References 2 and 3). For these accident occurrences, the presence of soluble boron in the spent fuel storage pit (controlled by LCO 3.7.15, "Spent Fuel Pit Boron Concentration") prevents criticality in both regions. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the auspices of the accompanying LCO.

The configuration of fuel assemblies in the fuel storage pit satisfies Criterion 2 of 10 CFR 50.36.

(continued)

BASES

LCO

Fuel assemblies stored in the spent fuel pit are classified in accordance with Figure 3.7.16-1 based on initial enrichment and burnup which is indicative of fuel assembly reactivity. Based on this classification, fuel assembly storage location within the spent fuel pit and storage location relative to other assemblies is restricted in accordance with the rules established by this LCO.

Fuel assemblies with an initial enrichment > 5.0 wt% U-235 are not shown on Figure 3.7.16-1 because fuel assemblies with this enrichment cannot be stored in the spent fuel pit in accordance with limits established in Technical Specification Section 4.3.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel pit.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in the spent fuel pit is not in accordance with this LCO, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with this LCO.

If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly in each location is in accordance with the accompanying LCO.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

2. SER related to Amendment 173 to Facility Operating License No. DPR-64, Indian Point Nuclear Generating Unit No. 3, April 15, 1997.

3. Criticality Analysis of the Indian Point 3 Fresh and Spent Fuel Racks, Westinghouse Commercial Nuclear Fuel Division, October, 1996.

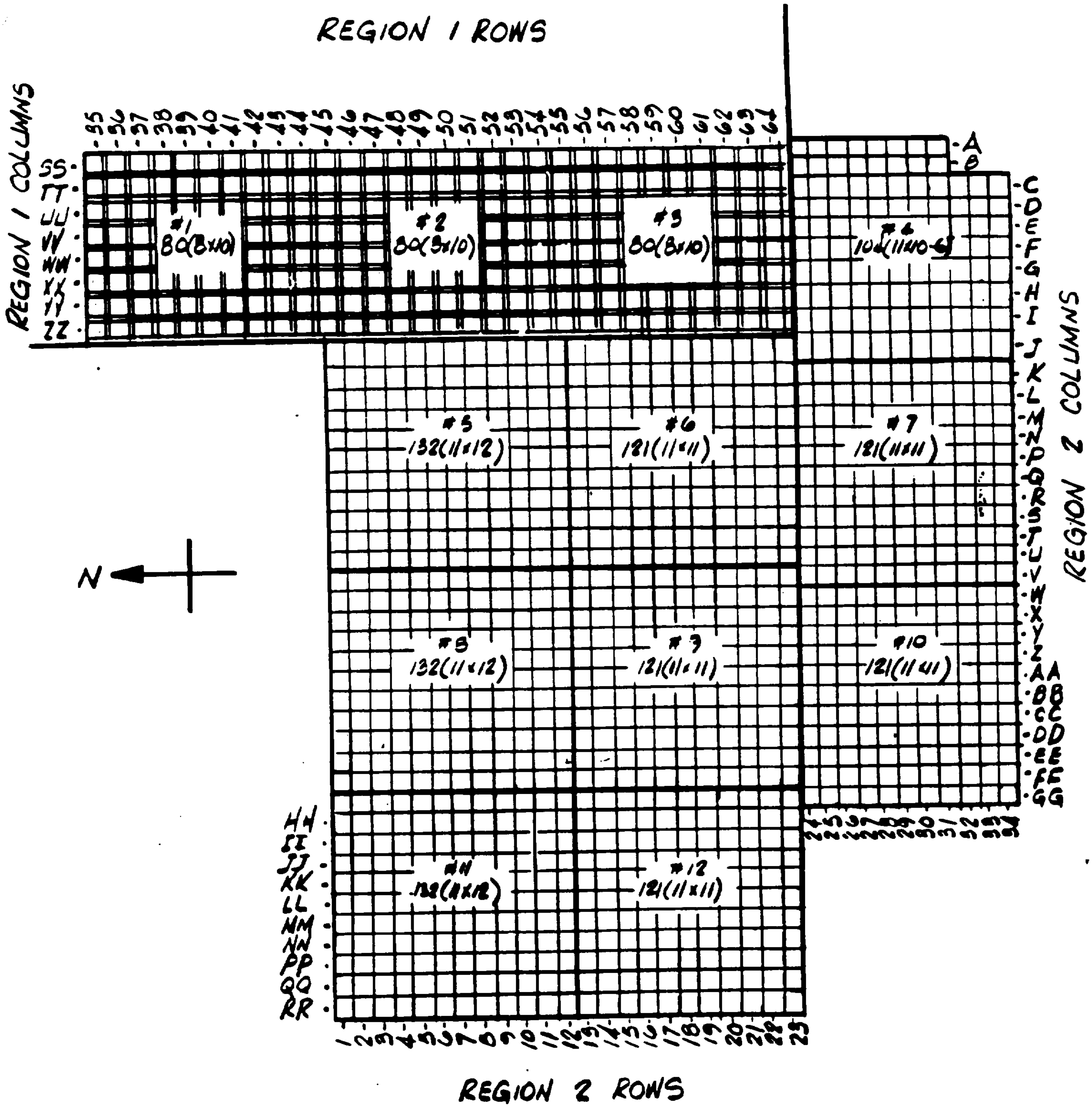

Figure B 3.7.16-1 (Page 1 of 1)
Maximum Density Spent Fuel Pit (SFP)
Racks, Regions and Indexing

B 3.7 PLANT SYSTEMS

B 3.7.17 Secondary Specific Activity

BASES

BACKGROUND

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

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

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours).

Operating a unit at the allowable limits could result in a 2 hour exclusion area boundary (EAB) or site boundary exposure of a small fraction (i.e., 10%) of the 10 CFR 100 (Ref. 1) limits (i.e., 25 rem whole body and 300 rem thyroid), or the limits established as the NRC staff approved licensing basis.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the FSAR, Chapter 14.2 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the EAB (i.e., site boundary) limits (Ref. 1) for whole body and thyroid dose rates.

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

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 assumed to discharge steam and any entrained activity through the MSSVs and ADVs during the event. Credit is taken in the analysis for activity plateout or retention; however, 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.

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 the required limit (Ref. 1).

(continued)

BASES

LCO (continued)	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, the steam generators are not normally used for heat removal. Both the RCS and steam generators 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 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 3.7.17.1

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,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1 (continued)

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 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 100.11.

2. FSAR, Chapter 14.2.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources – Operating

BASES

BACKGROUND

The unit Electrical Power Distribution System AC sources consist of the following: two offsite circuits (the normal or 138 kV circuit and the alternate or 13.8 kV circuit), each of which has a preferred and backup feeder; and, the onsite standby power circuit consisting of three diesel generators. As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The onsite plant distribution system is configured around 6.9 kV buses Nos. 1, 2, 3, 4, 5, and 6. All offsite power to safeguards buses enter the plant via 6.9 kV buses Nos.5 and 6 which are connected to the 138 kV (normal) offsite circuit and have the ability to be connected to the 13.8 kV (alternate) offsite circuit. 6.9 kV buses 1, 2, 3, and 4, which supply power to the 4 reactor coolant pumps (RCPs), typically receive power from the main generator via the unit auxiliary transformer (UAT) when the plant is at power. However, when the main generator or UAT is not capable of supporting this arrangement, 6.9 kV buses 1 and 2 receive offsite power via 6.9 kV bus 5 and 6.9 kV buses 3 and 4 receive offsite power via 6.9 kV bus 6. Following a unit trip, 6.9 kV buses 1, 2, 3, and 4 will auto transfer (fast transfer) to 6.9 kV buses 5 and 6 in order to receive offsite power. The 6.9 kV buses supply power to the 480 V buses using 6.9 kV/480 V station service transformers (SSTs) as follows: 6.9 kV bus 5 supplies 480 V bus 5A via SST 5; 6.9 kV bus 6 supplies 480 V bus 6A via SST 6; 6.9 kV bus 2 supplies 480 V bus 2A via SST 2; and, 6.9 kV bus 3 supplies 480 V bus 3A via SST 3.

The onsite AC Power Distribution System begins with 480 V buses 5A, 6A, 2A and 3A and is divided into 3 safeguards power trains (trains) consisting of the 480 volt safeguards bus(es) and associated AC electrical power distribution subsystems, 125 volt DC bus subsystems, and 120 volt vital AC instrument bus

(continued)

BASES

BACKGROUND (continued)

subsystems. The three trains are designed such that any two trains are capable of meeting minimum requirements for accident mitigation and/or safe shutdown. The three safeguards power trains are train 5A (480 volt bus 5A and associated DG 33), train 6A (480 volt bus 6A and associated DG 32), and train 2A/3A (480 volt buses 2A and 3A and associated DG 31).

Offsite power is supplied to the plant from the transmission network by two electrically and physically separated circuits, the 138 kV or normal circuit and the 13.8 kV or alternate circuit. Each of the offsite circuits from the Buchanan substation into the plant is required to be supported by a physically independent circuit from the offsite network into the Buchanan substation. All offsite power enters the plant via 6.9 kV buses Nos.5 and 6 which are connected to the 138 kV (normal) offsite circuit and have the ability to be connected to the 13.8 kV (alternate) offsite circuit. This arrangement satisfies the requirement that at least one of the two required circuits can within a few seconds, provide power to safety-related equipment following a loss-of-coolant accident. Operator action is required to supply offsite power to the plant using the 13.8 kV (alternate) offsite source.

The 138 kV circuit and the 13.8 kV circuit each have a preferred and a backup feeder that connects the circuit to the Buchanan substation. For both the 138 kV and 13.8 kV circuits, the preferred IP3 feeder is the backup IP2 feeder and the backup IP3 feeder is the preferred IP2 feeder.

For the 138 kV (i.e., normal) offsite circuit, IP2 and IP3 each have a dedicated Station Auxiliary Transformer (SAT) that can be supplied by either a preferred or backup feeder. The normal or 138 kV offsite circuit, including the SAT used exclusively for IP3, is designed to supply all IP3 loads, including 4 operating RCPs and ESF loads, when using either the preferred (95331) or backup (95332) feeder. There are no special restrictions when IP2 and IP3 are both using the same 138 kV feeder concurrently.

For the 13.8 kV (i.e., alternate) offsite circuit, there is a 13.8 kV/6.9 kV auto-transformer associated with feeder 13W92 and a 13.8 kV/6.9 kV auto-transformer associated with feeder 13W93.

(continued)

BASES

BACKGROUND (continued)

Feeder 13W93 and its associated auto-transformer is the preferred feeder for the IP3 alternate (13.8 kV) circuit and the backup feeder for the IP2 alternate (13.8 kV) circuit. Feeder 13W92 and its associated auto-transformer is the backup feeder for the IP3 alternate (13.8 kV) circuit and the preferred feeder for the IP2 alternate (13.8 kV) circuit.

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite 480 V ESF bus(es).

The onsite standby power source consists of 3 480 V diesel generators (DGs) with a separate DG dedicated to each of the safeguards power trains. Safeguards power train 5A (480 V bus 5A) is supported by DG 33; safeguards power train 6A (480 V bus 6A) is supported by DG 32; and, safeguards power train 2A/3A (480 V buses 2A and 3A) is supported by DG 31. A DG starts automatically on a safety injection (SI) signal or on an ESF bus undervoltage signal (refer to LCO 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation"). After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage, independent of or coincident with an SI signal. The DGs will also start and operate in the standby mode without tying to the ESF bus on an SI signal alone. Following the trip of offsite power, an undervoltage signal strips nonpermanent loads from the ESF bus. When the DG is tied to the ESF bus, loads are then sequentially connected to its respective ESF bus by individual load timers. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading the DG by automatic load application.

In the event of a loss of 138 kV or normal offsite source, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a loss of coolant accident (LOCA).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in

(continued)

BASES

BACKGROUND (continued)

the process. Within 1 minute after the initiating signal is received, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for DGs 31, 32 and 33 are consistent with the requirements of Regulatory Guide 1.9 (Ref. 3). The 3 DGs each consist of an Alco model 16-251-E engine coupled to a Westinghouse 2188 kVA, 0.8 power factor, 900 rpm, 3 phase, 60 cycle, 480 volt generator. The ESF loads that are powered from the 480 V ESF buses are listed in Reference 2.

The EDGs have four capacity ratings as defined below that can be used to assess EDG operability.

Continuous:	Electrical power output capability that can be maintained 24 hours /day, with no time constraint.
2000-hour:	Electrical power output capability that can be maintained in one continuous run of 2000 hours or in multiple shorter duration runs totaling 2000 hours.
2-hour:	Electrical power output capability that can be maintained for up to 2 hours in any 24-hour period.
1/2 - hour:	Electrical power output capability that can be maintained for up to 30 minutes in any 24-hour period.

The electrical output capabilities (EDG load) applicable to these four ratings are as follows:

RATING	EDG LOAD	TIME CONSTRAINT
Continuous	≤ 1750 kW	None
2000-hour	≤ 1950 kW	≤ 2000 hours / calendar year

(continued)

BASES

BACKGROUND (continued)

2-hour	≤ 1950 kW	≤ 2 hours in a 24-hour period; AND
	≤ 1750 kW	for the remaining 22 hours. [See NOTE A]
1/2-hour	≤ 2000 kW	≤ 30 minutes in a 24-hour period; AND
	≤ 1750 kW	for the remaining 23.5 hours. [See NOTE A]

NOTE A: The loading cycle permitted for the '2-hour' and the '1/2-hour' rating is operation at the overload condition (e.g. > 1750 kW) for the specified time followed by operation at the 'continuous' (e.g. ≤ 1750kW) rating for the remaining time in the 24-hour period. This loading cycle may be repeated each day, as long as back-to-back operation in the overload condition does not occur. The 2000-hour cumulative time constraint also applies to repetitive operation at the overload conditions allowed by the 2-hour and the 1/2-hour ratings.

Operation in excess of 2000 kW, regardless of the duration, is an unanalyzed condition. In such cases, the EDG is assumed to be inoperable and the vendor should be consulted to determine if accelerated or supplemental inspection and/or maintenance is necessary. The EDG can be returned to an operable status following completion of vendor-required inspection and/or maintenance.

APPLICABLE SAFETY ANALYSES

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Distribution Limits; 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the Accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least 2 of the 3 safeguards power trains energized from either onsite or offsite AC sources during Accident conditions in the event of:

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

b. A worst case single failure.

The AC sources satisfy Criterion 3 of 10 CFR 50.36.

LCO

Two qualified circuits between the offsite transmission network and the onsite Electrical Power System and separate and independent DGs for each train ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence (AOO) or a postulated DBA.

There are two qualified circuits (normal and alternate) from the transmission network at the Buchanan Station to the onsite electric distribution system. The normal circuit is 138 kV and the alternate circuit is 13.8 kV. If the alternate circuit is in use, the normal circuit is inoperable because the autotransfer functions mentioned in the following circuit descriptions are disabled. Both of these circuits must be supported by a circuit from the offsite network into the Buchanan substation that is physically independent from the other circuit to the extent practical. The circuits into the Buchanan substation that satisfy these requirements are 96951, 96952 and 95891.

(continued)

BASES

LCO (continued)

The 138 kV (i.e., normal) offsite circuit consists of one of the following: 138 kV feeder 95331 (preferred); or, 138 kV feeder 95332 (backup). Additionally, the 138 kV/6.9 kV station auxiliary transformer, circuit breakers ST5 and ST6 which supply 6.9 kV buses 5 and 6, and the following components which are common to the normal and alternate offsite circuits:

a. The 480 V bus 5A supply consisting of 6.9 kV bus 5, station service transformer 5, and circuit breakers SS5 and 52/5A;

b. The 480 V bus 2A supply consisting of 6.9 kV bus 5, circuit breaker UT2-ST5 (including autotransfer function), 6.9 kV bus 2, station service transformer 2, and circuit breakers SS2 and 52/2A;

c. The 480 V bus 6A supply consisting of 6.9 kV bus 6, station service transformer 6, and circuit breakers SS6 and 52/6A; and,

d. The 480 V bus 3A supply consisting of 6.9 kV bus 6, circuit breaker UT3-ST6 (including autotransfer function), 6.9 kV bus 3, station service transformer 3, and circuit breakers SS3 and 52/3A.

The 13.8 kV (i.e., alternate) offsite circuit consists of one ofthe following: 13.8 kV feeder 13W93 and its associated 13.8/6.9 kV autotransformer (preferred); or, 13.8 kV feeder 13W92 and its associated 13.8/6.9 kV autotransformer (backup). Circuit breakers GT35 and GT36, which supply 6.9 kV buses 5 and 6, and the following components are common to the normal and alternate offsite circuits:

a. The 480 V bus 5A supply consisting of 6.9 kV bus 5, station service transformer 5, and circuit breakers SS5 and 52/5A;

b. The 480 V bus 2A supply consisting of 6.9 kV bus 5, circuit breaker UT2-ST5 (not including autotransfer function), 6.9 kV bus 2, station service transformer 2, and circuit breakers SS2 and 52/2A;

(continued)

BASES

LCO (continued)

c. The 480 V bus 6A supply consisting of 6.9 kV bus 6, station service transformer 6, and circuit breakers SS6 and 52/6A; and,

d. The 480 V bus 3A supply consisting of 6.9 kV bus 6, circuit breaker UT3-ST6 (not including autotransfer function), 6.9 kV bus 3, station service transformer 3, and circuit breakers SS3 and 52/3A.

If the alternate (13.8 kV) offsite circuit is being used to supply power to the plant and the Unit Auxiliary Transformer is supplying 6.9 kV bus 1, 2, 3 or 4, the size of the 13.8 kV/6.9 kV auto-transformers requires that the automatic transfer of 6.9 kV buses 1, 2, 3, and 4 to 6.9 kV buses 5 and 6 (i.e., the offsite circuit) be disabled because neither 13.8 kV/6.9 kV auto-transformer is capable of supplying 4 operating RCPs. This requirement is not intended to preclude supplying 6.9 kV buses 1, 2, 3, and 4 using the alternate offsite circuit via the 13.8 kV/6.9 kV auto-transformers once sufficient loads have been stripped from 6.9 kV buses 1, 2, 3, and 4 to assure that the 13.8 kV/6.9 kV auto-transformer will not be overloaded by these manual actions.

If IP3 and IP2 are both using a single 13.8 kV feeder (13W92 or 13W93), administrative controls are used to ensure that the 13.8 kV/6.9 kV auto-transformer load restrictions will not be exceeded.

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ESF buses.

Three DGs must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This will be accomplished within 10 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses.

(continued)

BASES

LCO (continued)

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY.

The AC sources in each safeguards power train must be separate and independent (to the extent possible) of the AC sources in the other train. For the DGs, separation and independence are complete.

For the offsite AC sources, separation and independence are to the extent practical. A circuit may be connected to more than one ESF bus, and not violate separation criteria. A circuit that is not connected to an ESF bus is required to have OPERABLE automatic or manual transfer capability to the ESF buses to support OPERABILITY of that circuit.

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

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

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

The AC power requirements for MODES 5 and 6 are covered in LCO 3.8.2, "AC Sources - Shutdown."

ACTIONS

A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit on a more frequent basis. The LCO Bases describes the components and features which comprise the offsite circuits. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action not met.

(continued)

BASES

ACTIONS

A.1 (continued)

However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

A.2

Required Action A.2, applies only if the 13.8 kV offsite power circuit is being used to feed 6.9 kV buses 5 and 6 and the UAT is supplying 6.9 kV bus 1, 2, 3 or 4. This action prevents the automatic transfer of 6.9 kV buses 1, 2, 3, and 4 from the UAT to offsite power after a unit trip. Transfer of buses 1, 2, 3, and 4 from the UAT to offsite power could result in overloading the 13.8 kV/6.9 kV autotransformer. This requirement is not intended to preclude supplying 6.9 kV buses 1, 2, 3, and 4 using the alternate offsite circuit via the 13.8 kV/6.9 kV auto-transformers once sufficient loads have been stripped from 6.9 kV buses 1, 2, 3, and 4 to assure that the 13.8 kV/6.9 kV auto-transformer will not be overloaded by these manual actions. Automatic transfer of buses 1, 2, 3, and 4 can be disabled by placing 6.9 kV bus tie breaker control switches 1-5, 2-5, 3-6, and 4-6 in the "pull-out" position.

Although the auto-transfer feature is normally disabled prior to placing the 13.8 kV offsite power circuit in service, a Completion Time of 1 hour ensures that the 13.8 kV circuit meets requirements for Operability promptly when the alternate offsite circuit is configured to support the response of ESF functions.

A.3

Required Action A.3, which only applies if the train will not be powered automatically from an offsite source when the main turbine generator trips, is intended to provide assurance that an event coincident with a single failure of the associated DG will not result in a complete loss of redundant required features. Required safety features are designed with a redundant safety feature that is powered from a different safeguards power train.

(continued)

BASES

ACTIONS

A.3 (continued)

Therefore, if a required safety feature is supported by an inoperable offsite circuit, then the failure of the DG associated with that required safety feature will not result in the loss of a safety function because the safety function will be accomplished by the redundant safety feature that is powered from a different safeguards power train. However, if a required safety feature is supported by an inoperable offsite circuit and the redundant safety feature that is powered from a different safeguards power train is also inoperable, then the failure of the DG associated with that required safety feature will result in the loss of a safety function. Required Action A.3 ensures that appropriate compensatory measures are taken for a Condition where the loss of a DG could result in the loss of a safety function when an offsite circuit is not OPERABLE.

The turbine driven auxiliary feedwater pump is not required to be considered a redundant required feature, and, therefore, not required to be determined OPERABLE by this Required Action, because the design is such that the remaining OPERABLE motor driven auxiliary feedwater pump(s) is capable (without any reliance on the motor driven auxiliary feedwater pump powered by the emergency bus associated with the inoperable diesel generator) of providing 100% of the auxiliary feedwater flow assumed in the safety analysis.

The Completion Time for Required Action A.3 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The train will not have offsite power automatically supplying its loads following a trip of the main turbine generator; and

b. A required feature powered from another safeguards power train is inoperable.

(continued)

BASES

ACTIONS

A.3 (continued)

If at any time during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering that offsite power is not automatically available to one train of the onsite Electrical Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with the other train that has offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the two remaining safeguards power trains of the onsite Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.4

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Distribution System.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS (continued)

B.1

To ensure a highly reliable power source remains with an inoperable DG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of redundant required features. Required safety features are designed with a redundant safety feature that is powered from a different safeguards power train. Therefore, if a required safety feature is supported by an inoperable DG, then the failure of the offsite circuit will not result in the loss of a safety function because the safety function will be accomplished by the redundant safety feature that is powered from a different safeguards power train (and DG). However, if a required safety feature is supported by an inoperable DG and the redundant safety feature that is powered from a different safeguards power train is also inoperable, then a loss of offsite power will result in the loss of a safety function. Required Action B.2 ensures that appropriate compensatory measures are taken for a Condition where the loss of offsite power could result in the loss of a safety function when a DG is not OPERABLE.

The turbine driven auxiliary feedwater pump is not required to be considered a redundant required feature, and, therefore, not required to be determined OPERABLE by this Required Action, because the design is such that the remaining OPERABLE motor driven auxiliary feedwater pumps is capable (without any reliance on the motor driven auxiliary feedwater pump powered by the emergency bus associated with the inoperable diesel generator) of providing 100% of the auxiliary feedwater flow assumed in the safety analysis.

(continued)

BASES

ACTIONS

B.2 (continued)

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable DG exists; and

b. A required feature powered from another safeguards power train is inoperable.

If at any time during the existence of this Condition (one DG inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

Discovering one required DG inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with either OPERABLE DG, results in starting the Completion Time for the Required Action. A COMPLETION TIME of four hours from the discovery of these events existing concurrently is Acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

In this Condition, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS (continued)

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DG(s). If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DGs, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other DG(s), the other DG(s) would be declared inoperable upon discovery and Condition E of LCO 3.8.1 would be entered. Once the failure is repaired, the common cause failure no longer exists, and Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG(s), performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is reasonable to confirm that the OPERABLE DGs are not affected by the same problem as the inoperable DG.

B.4

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition B for a period that should not exceed 72 hours.

In Condition B, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Distribution System. The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Required Action C.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. Two offsite circuits are inoperable when both the immediate access circuit and the delayed offsite circuit are not available to one or more safeguards power trains. The most probable cause is a failure in a portion of the circuit that is common to both offsite circuits. The Completion Time for this failure of redundant required features is reduced to 12 hours from that allowed for one train without offsite power (Required Action A.3). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that three complete safeguards power trains are OPERABLE. When a redundant required feature is not OPERABLE, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains. This includes motor driven auxiliary feedwater pumps. Single train features, such as turbine driven auxiliary pumps, are included as discussed in the Bases for Required Action A.3. The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock."

In this Required Action the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable; and

b. A required feature is inoperable.

If at any time during the existence of Condition C (two offsite circuits inoperable) a required feature becomes inoperable, this Completion Time begins to be tracked.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition C for a period that should not exceed 24 hours.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more DGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient.

In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

According to Reference 6, with the available offsite AC sources, two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable, resulting in de-energization. When the UAT is being used to supply 6.9 kV buses 1, 2, 3 Or 4 and the 13.8 kV offsite circuit is being used to supply 6.9 kV buses 5 and 6, the autotransfer function is disabled. Therefore, 480 V safeguards buses 2A and 3A (safeguards train 2A/3A) will not be automatically re-energized with offsite power following a plant trip until connected to the offsite circuit by operator action. Therefore, the Required Actions of Condition D are modified by a Note to indicate that when Condition D is entered with no offsite or DG AC power source automatically available to any train, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems – Operating," must be immediately entered. This allows Condition D to provide requirements for the loss of one offsite circuit and one DG, without regard to whether a train would be de-energized during an event. LCO 3.8.9 provides the appropriate restrictions for a train that would be de-energized.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition D for a period that should not exceed 12 hours.

In Condition D, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition C (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure.

The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS (continued)

E.1

With two or more DGs inoperable, the remaining standby AC sources are not adequate to satisfy analysis assumptions. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to Reference 6, with two or more DGs inoperable, operation may continue for a period that should not exceed 2 hours.

F.1 and F.2

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

G.1 and H.1

Conditions G and H correspond to a level of degradation in which all redundancy in the AC electrical power supplies has been lost or a loss of safety function has already occurred. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. 1). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3),and Regulatory Guide 1.137 (Ref. 8).

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady state output voltage of 422 V is the value determined to be acceptable in the analysis of the degraded grid condition. This value allows for voltage drop to the terminals of 480 V motors. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating.

The specified maximum steady state output voltage of 500 V is equal to the maximum operating voltage specified for 480 V circuit breakers. The specified minimum and maximum frequencies of the DG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to ± 2% of the 60 Hz nominal frequency and are derived from the recommendations given in Regulatory Guide 1.9 (Ref. 3).

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. Portions of this SR are satisfied by telephone communication with Consolidated Edison personnel capable of confirming the status of the offsite circuits. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because 6.9 kV bus status and 13.8 kV circuit status are displayed in the control room.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.2

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note to indicate that all DG starts for these Surveillances may be preceded by an engine prelube period.

For the purposes of SR 3.8.1.2, the DGs are started from standby conditions. Standby conditions for a DG mean that the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations.

SR 3.8.1.2 requires that, at a 31 day Frequency, the DG starts from standby conditions and achieves required voltage and frequency within 10 seconds. The 10 second start requirement supports the assumptions of the design basis LOCA analysis in the FSAR, Chapter 14 (Ref. 5).

The normal 31 day Frequency for SR 3.8.1.2 is consistent with Regulatory Guide 1.9 (Ref. 3). This Frequency provides adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing. DGs have redundant air start motors and both air start motors are actuated by both channels of the start logic. The DG is OPERABLE when either air start motor is OPERABLE; however, this SR will not demonstrate that both of the air start motors are independently capable of starting the DG. If an air start motor is not capable of performing its intended function, a DG is inoperable until a timed start is conducted using the remaining air start motor. Alternately, this SR may be performed using one air start motor (i.e., redundant air start motor isolated) on a staggered basis to ensure that the DG will start with either air start motor.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads approximating the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while the 1.0 is an operational limitation to ensure circulating currents are minimized. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

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

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients, because of changing bus loads, do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test. Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank is at or above the level at which fuel oil is automatically added. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for approximately 1 hour of DG operation at full load.

The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and facility operators would be aware of any large uses of fuel oil during this period.

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day tanks once every 31 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are consistent with Regulatory Guide 1.137 (Ref. 8). This SR is for preventative maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during the performance of this Surveillance.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE.

The design of fuel transfer systems is such that pumps operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day tanks during or following DG testing. Therefore, a 31 day Frequency is appropriate. Since proper operation of fuel transfer systems is an inherent part of DG OPERABILITY, the Frequency of this SR is consistent with the 31 day Frequency for verification of DG operability.

SR 3.8.1.7

Transfer of the offsite power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. The 24 month Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note. The reason for the Note is that, during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and unit safety systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.8

Verification that 6.9 kV buses 2 and 3 will auto transfer (fast transfer) from the Unit Auxiliary transformer to 6.9 kV buses 5 and 6 (i.e. station auxiliary transformer) following a loss of voltage on 6.9 kV buses 2 and 3 is needed to confirm the Operability of a function assumed to operate to provide offsite power to safeguards power train 2A/3A following a trip of the main generator.

An actual demonstration of this feature requires the tripping of the main generator while the reactor is at power with the main generator supplying 6.9 kV buses 2 and 3. This will cause perturbations to the electrical distribution systems that could challenge unit safety systems during a plant shutdown. Therefore, in lieu of actually initiating a circuit transfer, testing that adequately shows the capability of the transfer is acceptable. This transfer testing may include any sequence of sequential, overlapping, or total steps so that the entire transfer sequence is verified. The 24 month Frequency is based on engineering judgement taking into consideration the plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle length.

This SR is modified by two Notes. The reason for Note 1 is that, during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge unit safety systems. Credit may be taken for unplanned events that satisfy this SR. As stated in Note 2, this SR is only required to be met when the 138 kV offsite circuit is supplying 6.9 kV buses 5 and 6 because, if the 13.8 kV circuit is supplying 6.9 kV buses 5 and 6, then the feature tested by this SR is required to be disabled.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.9

This Surveillance demonstrates that DG noncritical protective functions are bypassed on a loss of voltage signal concurrent with an ESF actuation test signal, and critical protective functions (engine overspeed, low lube oil pressure, and engine overcrank) trip the DG to avert substantial damage to the DG unit. The noncritical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

The 24 month Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DG from service.

SR 3.8.1.10

IEEE-387-1995 (Ref. 9) requires demonstration once per 24 months that the DGs can start and run continuously at full load capability for an interval of not less than 8 hours, ≥ 105 minutes of which is at a load equivalent to 110% of the continuous duty rating and the remainder of the time at a load equivalent to the continuous duty rating of the DG. The DG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.10 (continued)

In order to ensure that the DG is tested under load conditions that are as close to design conditions as possible, testing must be performed using a power factor of ≤ 0.9. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

The 24 month Frequency is consistent with the recommendations of Ref. 9, and takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This Surveillance is modified by two Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the power factor limit will not invalidate the test. The reason for Note 2 is that, during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and unit safety systems.

SR 3.8.1.11

Under accident conditions with concurrent loss of offsite power, loads are sequentially connected to the bus by individual load timers to prevent overloading of the DGs due to high motor starting currents. The design load sequence time interval tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 2 provides a summary of the automatic loading of ESF buses.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

The Frequency of 18 months is based on engineering judgment, taking into consideration operating experience that has shown that these components usually pass the SR. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that specifies that load timers associated with equipment that has automatic initiation capability disabled are not required to be OPERABLE. This note is needed because these time delay relays affect the OPERABILITY of both the AC sources (offsite power and DG) and the specific load that the relay starts. If a timer fails to start a required load or starts the load later than assumed in the analysis, then the required load is not OPERABLE. If a timer starts the load outside the design interval (early or late), then the DG and offsite source are not OPERABLE because overlap of equipment starts may cause an offsite source to exceed limits for voltage or current or a DG to exceed limits for voltage, current or frequency. Therefore, when an individual load sequence timer is not OPERABLE, because the timing sequence is outside the design interval, Condition D must be entered. However, if the automatic initiation capability of the affected load is disabled, Condition D may be exited, and the Actions for the inoperable load are taken. It is conservative to disable the automatic initiation capability of a component rather than continue with the associated DG inoperable because of the following: the potential for adverse impact on the DG by simultaneous start of ESF equipment is eliminated; all other loads powered from the safeguards power train are available to respond to the event; and, the load with the inoperable timer remains available for a manual start after the one minute completion of the normal starting sequence.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.12

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates the DG operation during a loss of offsite power actuation test signal in conjunction with an ESF actuation signal. This SR verifies all actions encountered from an ESF signal concurrent with the loss of offsite power, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency within the specified time.

The DG autostart time of 10 seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability is achieved.

The requirement to verify the connection and power supply of permanent and auto-connected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, or high pressure injection systems are not capable of being operated, or residual heat removal (RHR) systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation.

In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.12 (continued)

The Frequency of 24 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 24 months.

This SR is modified by three Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil and temperature maintained and lube oil continuously circulated consistent with manufacturer recommendations for DGs.

The reason for Note 2 is that the performance of the Surveillance would remove required offsite circuits from service, perturb the electrical distribution system, and challenge safety systems.

The reason for Note 3 is to allow the SR to be conducted with only one safeguards train at a time or with two or three safeguards trains concurrently. Allowing the LOOP/LOCA test to be conducted using one safeguards power train and one DG at a time is acceptable because the safeguards power trains are designed to respond to this event independently. Therefore, an individual test for each safeguards power train will provide an adequate verification of plant response to this event.

Simultaneous testing of all three safeguards power trains is acceptable as long as the following plant conditions are established:

- All three DGs are available,
- diverse and redundant decay heat removal is available,
- no offsite power circuits are inoperable, and
- no simultaneous activities are performed that are precursors to events requiring AC power for mitigation (e.g., fuel handling accident or inadvertent RCS draindown)

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.13

This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the DGs are started simultaneously.

The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3).

This SR is modified by two Notes. The reason for Note 1 is to minimize wear on the DG during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations.

The reason for Note 2 is to allow SR 3.8.1.12 to satisfy the requirements of this SR if SR 3.8.1.12 is performed with more than one safeguards power train concurrently.

REFERENCES

1. 10 CFR 50, Appendix A.
2. FSAR, Chapter 8.
3. Regulatory Guide 1.9, Rev. 3, July 1993.
4. FSAR, Chapter 6.
5. FSAR, Chapter 14.
6. Regulatory Guide 1.93, Rev. 0, December 1974.
7. Generic Letter 84-15, Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability.
8. Regulatory Guide 1.137, Rev. 0, 1978.

(continued)

BASES

REFERENCES (continued)	9.	IEEE Standard 387-1995, IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

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

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

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

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

In the event of an accident during shutdown, this LCO ensures the capability to support systems necessary to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC sources satisfy Criterion 3 of 10 CFR 50.36.

(continued)

BASES (continued)

LCO

One offsite circuit capable of supplying the onsite power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems – Shutdown," ensures that all required loads are powered from offsite power. Two OPERABLE DGs, associated with the distribution system train required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and DGs ensures the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents). Under specific plant conditions the number of required operable DGs may be reduced to one. The plant conditions described by the LCO ensures that ample time is available for operator actions in response to a loss of offsite power.

The offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the Engineered Safety Feature (ESF) bus(es). Offsite circuits are those that are described in the Bases of LCO 3.8.1, AC Sources - Operating, except that safeguards power trains may be cross connected when in MODES 5 and 6.

The DGs must be capable of starting, accelerating to rated speed and voltage, and connecting to their respective ESF bus on detection of bus undervoltage. This sequence must be accomplished within 10 seconds. The DG must be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY.

It is acceptable for safeguards power trains to be cross tied during shutdown conditions, allowing a single offsite power circuit to supply all required trains. In this case, interlocks that disconnect the affected tie breakers before DGs are automatically connected to the bus must be OPERABLE.

(continued)

BASES (continued)

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

ACTIONS

A.1

An offsite circuit would be considered inoperable if it were not available to one required safeguards power train. Although two safeguards power trains may be required by LCO 3.8.10, the one train with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By the allowance of the option to declare required features inoperable, with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

A.2.1, A.2.2, A.2.3 and A.2.4

With the offsite circuit not available to all required trains, the option would still exist to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently

(continued)

BASES

ACTIONS

A.2.1, A.2.2, A.2.3 and A.2.4 (continued)

conservative actions is made. It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions. The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SDM is maintained.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System's ACTIONS would not be entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no AC power to any required ESF bus, the ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit, whether or not a train is de-energized. LCO 3.8.10 would provide the appropriate restrictions for the situation involving a de-energized bus.

(continued)

BASES

ACTIONS (continued)

B.1, B.2, B.3 and B.4

Condition B is entered when any required DGs are inoperable. A DG would be considered inoperable if it could not support its associated safeguards power train. When LCO 3.8.2.b.1 applies, 2 DGs are required to be OPERABLE. In this case, whether one or both of the required DGs is inoperable, the minimum required diversity of AC power sources is not available to required features. Therefore, it is required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactive additions.

When specific limitations are satisfied, as stated in LCO 3.8.2.b.2, only one DG is required. The additional restrictions on plant conditions for requiring only one DG provides ample time for operator action, in the event of a loss of offsite power, to manually restore decay heat removal capability. The combination of subcritical duration, fuel location, and refueling cavity water level results in a time period of at least 3 hours for heatup of this water inventory from 140 ^{0}F to 200 ^{0}F.

With any required DGs inoperable, the Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SDM is maintained. Additionally, Required Actions B.1, B.2, and B.3 do not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability or the occurrence of postulated events.

Furthermore, when Required Actions B.1, B.2 and B.3 are implemented, it is required to immediately initiate action (B.4) to restore the required DG(s) and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.8 is not required to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.9 is not required to be met because the DG automatic trips are bypassed only on the safety injection start signal, not on the loss of power start signal. SR 3.8.1.13 is excepted because starting independence is not required with the DG(s) that is not required to be operable.

This SR is modified by two Notes. The reason for the first Note is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude deenergizing a required 480 V ESF bus or disconnecting a required offsite circuit during performance of SRs. With limited AC sources available, a single event could compromise both the required circuit and the DG. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

The reason for the second Note is that SR 3.8.1.12 includes testing with an actual or simulated ESF actuation signal. ESF actuation is not required in MODES 5 and 6 so that this portion of the surveillance is not required to be met.

REFERENCES None.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil and Starting Air

BASES

BACKGROUND

Fuel oil for the safeguards DGs is stored in three 7,700 gallon DG fuel oil storage tanks located on the south side of the Diesel Generator Building. The offsite DG fuel oil reserve is maintained in two 30,000 gallon tanks located in the Indian Point 1 Superheater Building and/or a 200,000 gallon tank in the Buchanan Substation which is located in close proximity to the IP3 site. The IP3 offsite fuel oil reserve is maintained by the operators of IP2, in accordance with formal agreements. The IP3 offsite DG fuel oil reserve is normally stored in the same tanks used to store the IP2 offsite DG fuel oil reserve.

Sufficient fuel for at least 48 hours of minimum safeguards equipment operation is available when any two of the DG fuel oil storage tanks are available and each contains 5,365 usable gallons of fuel oil. Additional margin is provided by 115 gallons of fuel oil in the DG day tank required by SR 3.8.1.4. The maximum DG loadings for design basis transients that actuate safety injection are summarized in FSAR 8.2 (Ref. 1). These transients include large and small break loss of coolant accidents (LOCA), main steamline break and steam generator tube rupture (SGTR).

The three DG fuel oil storage tanks are filled through a common fill line that is equipped with a truck hose connection and a shutoff valve at each tank. The overflow from any DG fuel oil storage tank will cascade into an adjacent tank. Each DG fuel oil storage tank is equipped with a single vertical fuel oil transfer pump that discharges to either the normal or emergency header. Either header can be used to fill the day tank at each diesel. Each DG fuel oil storage tank has an alarm that sounds in the control room when the level in the tank approaches the level equivalent of the minimum required usable inventory. Each tank is also equipped with a sounding connection and a level indicator.

(continued)

BASES

BACKGROUND (continued)

Each emergency diesel is equipped with a 175-gallon day tank with an operating level that provides sufficient fuel for approximately one hour of DG operation. A decrease in day tank level to approximately 115 gallons (65% full) will cause the normal and emergency fill valves on that day tank to open and the transfer pump in the corresponding DG fuel oil storage tank to start. Once started, the pump will continue to run until that day tank is filled. However, any operating transfer pump will fill any day tank with a normal or emergency fill valve that is open. When a day tank is at approximately 158 gallons (90% full), a switch initiates closing of the day tank normal and emergency fill valves.

Technical Specifications require sufficient fuel oil to operate 2 of the 3 required DGs at minimum safeguards load for 7 days. The Technical Specification required volume of fuel oil includes the 26,826 gallons of usable fuel oil in the reserve tanks, and 10,730 usable gallons in two DG fuel oil storage tanks (assuming a failure makes the oil in the third DG fuel oil storage tank unavailable), without crediting the additional margin of 230 gallons in two day tanks (assuming a failure makes the oil in the day tank associated with the third DG unavailable).

If the DGs require fuel oil from the fuel oil reserve tank(s), the fuel oil will be transported by truck to the DG fuel oil storage tanks. A truck with appropriate hose connections and capable of transporting oil is available either on site or at the Buchanan Substation. Commercial oil supplies and trucking facilities are also available in the vicinity of the plant.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. Requirements for DG fuel oil testing methodology, frequency, and acceptance criteria are maintained in the program required by Specification 5.5.12, Diesel Fuel Oil Testing Program.

Each DG has an air start system with adequate capacity for four successive start attempts on the DG without recharging the air start receiver(s). The air starting system is designed to shutdown and lock out any engine which does not start during the initial start attempt so that only enough air for one automatic start is used. This conserves air for subsequent DG start attempts.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 14 (Ref. 3), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

Since diesel fuel oil and the air start subsystem support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36.

LCO

Stored diesel fuel oil is required to have sufficient supply for 7 days of operation for 2 of 3 DGs at minimum safeguards load. Fuel oil is also required to meet specific standards for quality. This requirement, in conjunction with an ability to obtain replacement supplies within 7 days, supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an anticipated operational occurrence (AOO) or a postulated DBA with loss of offsite power. DG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources – Operating," and LCO 3.8.2, "AC Sources – Shutdown."

The starting air system is required to have a minimum capacity for four successive DG start attempts without recharging the air start receivers.

(continued)

BASES (continued)

APPLICABILITY

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

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

A.1

In this Condition, the requirements of SR 3.8.3.2.a are not met. Therefore, a DG will not be able to support 48 hours of continuous operation at minimum safeguards load and replenishment of the DG fuel oil storage tanks will be required in less than 48 hours following an accident. The DG associated with the DG fuel oil storage tank not within limits must be declared inoperable immediately because replenishment of the DG fuel oil storage tank requires that fuel be transported from the offsite DG fuel oil reserve by truck and the volume of fuel oil remaining in the DG fuel oil storage tank may not be sufficient to allow continuous DG operation while the fuel transfer is planned and conducted under accident conditions.

This Condition is preceded by a Note stating that Condition A is applicable only in MODES 1, 2, 3 and 4. This Note provides recognition that reduced DG loading required to respond to events in MODES 5 and 6 significantly reduces the amount of fuel oil required in the DG fuel oil storage tanks when in these MODES.

(continued)

BASES

ACTIONS (continued)

B.1

In this Condition, the requirements of SR 3.8.3.2.b are not met. With less than the total required minimum fuel oil in one or more DG fuel oil storage tanks, the one or two DGs required to be operable in MODES 5 and 6 and during movement of irradiated fuel may not have sufficient fuel oil to support continuous operation while a fuel transfer from the offsite DG fuel oil reserve or from another offsite source is planned and conducted under accident conditions. Fuel oil credited to meet this requirement must be in one or more storage tanks associated with the operable DG(s) because the fuel transfer pump in each tank may depend on power from that DG.

This condition requires that all DGs be declared inoperable immediately because minimum fuel oil level requirements in SR 3.8.3.2.b is a condition of Operability of all DGs when in the specified MODES.

This Condition is preceded by a Note stating that Condition B is applicable only in MODES 5 and 6 and during the movement of irradiated fuel. This Note provides recognition that reduced DG loading required to respond to events in MODES 5 and 6 significantly reduces the amount of fuel oil required in the DG fuel oil storage tanks when in these MODES.

C.1

In this Condition, the fuel oil remaining in the offsite DG fuel oil reserve is not sufficient to operate 2 of the 3 DGs at minimum safeguards load for 7 days. Therefore, all 3 DGs are declared inoperable immediately.

This Condition is preceded by a Note stating that Condition D is applicable only in MODES 1, 2, 3 and 4 because the offsite DG fuel oil reserve is required to be available only in these MODES. This Note provides recognition that reduced DG loading required to respond to events in MODES 5 and 6 significantly reduces the amount of fuel oil required when in these MODES.

(continued)

BASES

ACTIONS (continued)

D.1

This Condition is entered as a result of a failure to meet the acceptance criteria of SR 3.8.3.3 or SR 3.8.3.4 when the DG fuel oil storage tanks or reserve storage tanks are verified to have particulate within the allowable value in Specification 5.5.12, Diesel Fuel Oil Testing Program. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7-day and 30-day Completion Times, for the onsite tanks and the reserve storage tanks, respectively, allows for further evaluation, resampling and re-analysis of the DG fuel oil.

E.1

This condition is entered as a result of a failure to meet the acceptance criteria of SR 3.8.3.3 or SR 3.8.3.4 when the DG fuel oil storage tanks or reserve storage tanks are verified to have properties (other than particulates) within the allowable values of Specification 5.5.12, Diesel Fuel Oil Testing Program. A period of 30 days is allowed to restore the properties of the fuel oil in the DG fuel oil storage tank to within the limits established by Specification 5.5.12. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that

(continued)

BASES

ACTIONS

E.1. (continued)

the DG would still be capable of performing its intended function. A period of 60 days is allowed to restore the properties of the fuel oil stored in the affected reserve storage tank to within the limits established by Specification 5.5.12. This period provides sufficient time to perform the actions described above for the DG fuel oil storage tanks. The additional time allowed for the reserve tanks is acceptable because reserve oil is not immediately needed to support DG operation and reserve oil is available from more than one reserve tank. Reserve oil is also available from commercial suppliers in the vicinity of the plant.

F.1

With starting air receiver pressure < 250 psig, sufficient capacity for four successive DG start attempts does not exist. However, as long as the receiver pressure is ≥ 90 psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period. Entry into Condition F is not required when air receiver pressure is less than required limits while the DG is operating following a successful start.

G.1

With a Required Action and associated Completion Time not met, or one or more DG's fuel oil or starting air subsystem not within limits for reasons other than addressed by Conditions A through F, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the offsite DG fuel oil reserve to support 2 DGs at minimum safeguards load for 7 days assuming requirements for the DG fuel oil storage tanks and day tanks are met. The 7 day duration with 2 of the 3 DGs at minimum safeguards load is sufficient to place the unit in a safe shutdown condition and to bring in replenishment fuel from a commercial source.

The 24 hour Frequency is needed because the DG fuel oil reserve is stored in fuel oil tanks that support the operation of gas turbine peaking units that are not under IP3 control. Specifically, the 26,826 gallons needed to support 7 days of DG operation is maintained in two 30,000 gallon tanks located in the Indian Point 1 Superheater Building and/or a 200,000 gallon tank in the Buchanan Substation. Although the volume of fuel oil required to support IP3 DG operability is designated as for the exclusive use of IP3, the fact that the oil in the storage tanks is used for purposes other than IP3 DGs and oil consumption is not under the direct control of IP3 operators warrants frequent verification that required offsite DG fuel oil reserve volume is being maintained.

SR 3.8.3.2

SR 3.8.3.2.a provides verification when in MODES 1, 2, 3, and 4, that there is an adequate inventory of fuel oil in the storage DG fuel oil tanks to support each DG's operation for at least 48 hours of operation of minimum safeguards equipment when any two of the DG fuel oil storage tanks are available and 5,365 gallons of usable fuel oil is contained in each tank.

SR 3.8.3.2.b provides verification when in MODES 5 and 6 and during movement of irradiated fuel that the minimum required fuel oil for operation in these MODES is available in one or more DG fuel oil storage tanks. The minimum required volume of fuel oil

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.2 (continued)

takes into account the reduced DG loading required to respond to events in MODES 5 and 6 is sufficient to support the two DGs required to be operable in MODES 5 and 6 and during movement of irradiated fuel while a fuel transfer from the offsite DG fuel oil reserve or from another offsite source is planned and conducted under accident conditions.

This minimum volume required by SR 3.8.3.2.a and SR 3.8.3.2.b is the usable volume and does not include allowances for fuel not usable due to the fuel oil transfer pump cutoff switch (worst case 956 gallons for #33 tank and 915 gallons for #31 and #32 tanks) and margin (20 gallons per tank). If the installed level indicators are used to measure tank volume, an additional allowance of 50 gallons for instrument uncertainty associated with the level indicators must be included. Appropriate adjustments are required for SR 3.8.3.2.b if the required volume is found in more than one DG fuel oil storage tank.

The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.8.3.3

This surveillance verifies that the properties of new and stored fuel oil meet the acceptance criteria established by Specification 5.5.12, "Diesel Fuel Oil Testing Program." Specific sampling and testing requirements for diesel fuel oil in accordance with applicable ASTM Standards are specified in the administrative program developed to ensure Specification.

New fuel oil is sampled prior to addition to the DG fuel oil storage tanks and stored fuel oil is periodically sampled from the DG fuel oil storage tanks. Requirements and acceptance

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.3 (continued)

criteria for fuel oil are divided into 3 parts as follows: a) tests of the sample of new fuel sample and acceptance criteria that must be met prior to adding the new fuel to the DG fuel oil storage tanks; b) tests of the sample of new fuel that may be completed after the fuel is added to the DG fuel oil storage tanks; and, c) tests of the fuel oil stored in the DG fuel oil storage tanks. The basis for each of these tests is described below.

The tests of the sample of new fuel and acceptance criteria that must be met prior to adding the new fuel to the DG fuel oil storage tanks are a means of determining that the new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. The tests, limits, and applicable ASTM Standards needed to satisfy Specification 5.5.12 are listed in the administrative program developed to implement Specification 5.5.12.

Failure to meet any of the specified limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO because the fuel oil is not added to the storage tanks.

The tests of the sample of new fuel that may be completed after the fuel is added to the DG fuel oil storage tanks must be completed Within 31 days. The fuel oil is analyzed to establish that the other properties of the fuel oil meet the acceptance criteria of Specification 5.5.12. The 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.3 (continued)

effect on DG operation. Failure to meet the specified acceptance criteria requires entry into Condition E and restoration of the quality of the fuel oil in the DG fuel oil storage tank within the associated Completion Time and explained in the Bases for Condition E. This Surveillance ensures the availability of high quality fuel oil for the DGs.

The periodic tests of the fuel oil stored in the DG fuel oil storage tanks verify that the length of time or conditions of storage has not degraded the fuel in a manner that could impact DG OPERABILITY. Fuel oil degradation during long term storage shows up as an increase in particulate, due mostly to oxidation. The presence of particulate does not mean the fuel oil will not burn properly in a diesel engine. The particulate can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure. Particulate concentrations must meet the acceptance criteria of Specification 5.5.12. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. Each DG fuel oil storage tank must be considered and tested separately.

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.4

The IP3 offsite fuel oil reserve is maintained by the operators of IP2, in accordance with formal agreements. The IP3 offsite DG fuel oil reserve is normally stored in the same tanks used to store the IP2 offsite DG fuel oil reserve. Fuel oil properties of new and stored fuel are controlled in accordance with IP2 Technical Specifications and FSAR in order to meet requirements for the Operability of IP2 and IP3 DGs.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.4 (continued)

Required testing of the properties of new and stored fuel in the offsite DG fuel oil reserve is performed by IP2 in accordance with programs established by IP2. IP3 performs periodic verification that fuel oil stored in the offsite DG fuel oil reserve meet the requirements of Specification 5.5.12.

Failure to meet the specified acceptance criteria, whether identified by IP2 or IP3, requires entry into Condition D or E and restoration of the quality of the fuel oil in the offsite DG fuel oil reserve within the associated Completion Time and explained in the Bases for Conditions D and E.

SR 3.8.3.5

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of four engine starts without recharging. Failure of the engine to start within approximately 15 seconds indicates a malfunction at which point the overcrank relays terminate the start cycle. In this condition, sufficient starting air will still be available so that the DG can be manually started. The pressure specified in this SR is intended to reflect the lowest value at which the four starts can be accomplished.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.6

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks once every 92 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, and contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are consistent with Regulatory Guide 1.137 (Ref. 2). This SR is for preventive maintenance. Unless the volume of water is sufficient that it could impact DG OPERABILITY, presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed within 7 days of performance of the Surveillance.

REFERENCES

1. FSAR, Section 8.2.
2. Regulatory Guide 1.137.
3. FSAR, Chapter 14.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources – Operating

BASES

BACKGROUND

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and preferred 120 V AC vital instrument bus power (via inverters). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also is consistent with the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

The 125 VDC electrical power system consists of four independent safety related DC electrical power subsystems (31, 32, 33 and 34). Each subsystem consists of one 125 VDC battery, the associated battery charger for each battery (except that battery charger 34 is not covered by this LCO), and all the associated control equipment and interconnecting cabling. In addition, battery charger 35 is an installed spare that can be used as the associated charger for any one of the batteries (Ref. 4).

The four DC electrical power subsystems (batteries and associated chargers) 31, 32, 33, and 34 feed four main distribution power panels. DC electrical power subsystems 31, 32, and 33 supply DC control power to 480 volt buses Nos. 5A, 6A, and 2A/3A, respectively. The 480 volt switchgear bus sections that supply power to the safeguards equipment also receive DC control power from its associated DC electrical power subsystem. DC electrical power subsystem 34 does not provide DC control power to any equipment assumed to function to mitigate an accident.

The DC electrical power subsystems 31, 32, 33 and 34 also provide DC electrical power to the inverters, which in turn power the AC vital instrument buses. As a result, each of the four DC electrical power subsystems supports one of the four Reactor

(continued)

BASES

BACKGROUND (continued)

Protection System (RPS) Instrumentation channels and one of the four Engineered Safety Features Actuation (ESFAS) Instrumentation channels. DC electrical power subsystems 31 and 32 each support one of the two trains of RPS Instrumentation actuation logic and one of the two trains of ESFAS Instrumentation actuation logic. Electrical distribution, including DC Sources, is described in the FSAR (Ref. 4).

During normal operation, the 125 VDC load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC load is automatically powered from the station batteries.

Each of the four station batteries is sized to carry its expected shutdown loads for a period of 2 hours without battery terminal voltage falling below 105 volts following a plant trip that includes a loss of all AC power. Major loads with their approximate operating times on each battery are listed in Reference 4. The four battery chargers have been sized to recharge discharged batteries within 15 hours while carrying the normal DC subsystem load.

Battery 34 and charger 34 were installed in 1979 (along with inverter 34) to ensure a continuous power supply to 120 V AC vital instrument bus (VIB) 34 which supports RPS and ESFAS channel III. Prior to this modification, VIB 34 was powered solely by two 480 V/120 V constant voltage transformers (CVTs) supplied by separate safeguard power trains. Although these two CVTs provide redundant safety related power supplies for VIB 34, these power sources are unavailable following a loss of offsite power until the emergency diesel generators re-power one or both of the associated safeguards power trains. Additionally, battery 34 (via the associated inverter) provides a continuous power supply for VIB 34 which decreases the potential for an inadvertent reactor trip or ESFAS actuation, especially when an instrument channel associated with a different VIB is inoperable and in trip. Note that battery charger 34 is not required by LCO 3.8.4. This is acceptable because VIB 34 can be powered by either of the two CVTs supplied by separate safeguard power trains if battery charger 34 is not available following an event.

(continued)

BASES

BACKGROUND (continued)

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

Each 125 VDC battery is separately housed in a ventilated room apart from its charger and power panels. Each subsystem is separated electrically from the other subsystems to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. There is no sharing between redundant subsystems, such as batteries, battery chargers, or power panels.

The batteries are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of $\geq$ 123.5 V for batteries 31 and 32 (each consisting of 58 cells) and $\geq$ 127.8 V for batteries 33 and 34 (each consisting of 60 cells).

Each DC electrical power subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank charged as necessary to meet the requirements of LCO 3.8.6, Battery Parameters. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to the required charged state within 15 hours while supplying normal steady state loads discussed in the FSAR, Chapter 8 (Ref. 4).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 14 (Ref. 6), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power subsystems 31, 32 and 33 provide normal and emergency DC electrical power for the DGs, and control and switching during all MODES of operation. Each of the four DC electrical power subsystems supports one of the four 120 V AC vital instrument buses via an inverter.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power (i.e., emergency diesel generators); and

b. A worst case single failure.

The DC sources satisfy Criterion 3 of 10 CFR 50.36.

LCO

This LCO requires the OPERABILITY of the following four DC electrical power subsystems:

Battery 31 and associated Battery Charger;
Battery 32 and associated Battery Charger;
Battery 33 and associated Battery Charger; and
Battery 34.

In addition, the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the train are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Loss of any train DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE DC electrical power subsystem requires the battery and respective charger to be operating and connected to the associated DC bus.

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

(continued)

BASES

APPLICABILITY (continued)

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

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

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

ACTIONS

A.1

Condition A is entered when battery No. 34 is not OPERABLE. The only safety related load supported by DC subsystem 34 is 120 V AC vital instrument bus 34 which is supplied via inverter 34. Therefore, the Required Actions for inverter 34 not OPERABLE specified in LCO 3.8.7, Inverters-Operating, are appropriate when battery No. 34 is not OPERABLE. Additionally, ITS 3.8.9 (and ITS Section 3.3) ensure that 120 V AC vital instrument bus 34 is energized when required. The 2 hour Completion Time is consistent with the completion time for an inoperable battery and/or charger in any of the other three DC electrical power subsystems.

B.1

Condition B is entered when DC subsystem 31, 32 or 33 (battery and/or associated charger) is not Operable. Loss of DC subsystem 34 (Condition A) differs from the loss of DC subsystem 31, 32 or 33 (Condition B) because Condition B could result in the loss of DC control power to 480 volt bus No. 5A, 6A, or 2A/3A, respectively, and the associated emergency diesel generator. Therefore, this Condition represents a significant degradation of the ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation.

(continued)

BASES

ACTIONS

B.1 (continued)

It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential loss of additional DC subsystems.

If one of the required DC electrical power subsystems is inoperable (e.g., inoperable battery, inoperable battery charger, or inoperable battery charger and associated inoperable battery), the remaining DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure would, however, result in the loss of another 125 VDC electrical power subsystems with attendant loss of ESF functions, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 7) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

C.1 and C.2

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. The Completion Time to bring the unit to MODE 5 is consistent with the time required in Regulatory Guide 1.93 (Ref. 7).

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. The 31 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref.8).

SR 3.8.4.2

This SR requires that each battery charger be capable of supplying the voltage and current necessary to recharge partially discharged batteries (two hour discharge at a rate that does not cause battery terminal voltage to fall below 105 volts). These requirements are consistent with the output rating of the chargers (Ref. 4). Therefore, this SR can be satisfied by operating each charger at the design voltage and current for a minimum of 2 hours. According to Regulatory Guide 1.32 (Ref. 9), the battery charger supply is required to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied.

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.2 (continued)

performance during these 24 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

This Surveillance is required to be performed during MODES 5 and 6 since it would require the DC electrical power subsystem to be inoperable during performance of the test.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

SR 3.8.4.3

A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements as specified in Reference 4.

The Surveillance Frequency of 24 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 9) and Regulatory Guide 1.129 (Ref. 10), which state that the battery service test should be performed during refueling operations or at some other outage.

A modified performance discharge test may be performed in lieu of a service test.

The modified performance discharge test is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a rated one minute discharge represents a

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.3 (continued)

very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

A modified performance discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

SR 3.8.4.4

A battery performance discharge test is a test of constant current capacity of a battery, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

A battery modified performance discharge test is described in the Bases for SR 3.8.4.3. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.4; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.4 while satisfying the requirements of SR 3.8.4.3 at the same time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.4 (continued)

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 8) and IEEE-485 (Ref. 5). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity ≥ 100% of the manufacturer's rating. Degradation is indicated, according to IEEE-450 (Ref. 8), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ 10% below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 8).

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

SR 3.8.4.5

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

REFERENCES

1. 10 CFR 50, Appendix A.

2. Regulatory Guide 1.6, March 10, 1971.

(continued)

BASES

REFERENCES (continued)

3. IEEE-308-1978.

4. FSAR, Chapter 8.

5. IEEE-485-1983, June 1983.

6. FSAR, Chapter 14.

7. Regulatory Guide 1.93, December 1974.

8. IEEE-450-1995.

9. Regulatory Guide 1.32, February 1977.

10. Regulatory Guide 1.129, December 1974.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources – Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter 14 (Ref. 1), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

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

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

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The DC sources satisfy Criterion 3 of 10 CFR 50.36.

(continued)

BASES (continued)

LCO

The four DC electrical power subsystems, each subsystem consisting of one battery, one battery charger (except for battery charger 34 which is not covered by this LCO), and the corresponding control equipment and interconnecting cabling within the safeguards power train, are required to be OPERABLE to support required safeguards power trains of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems – Shutdown." This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

DC subsystems 31 and 32 may be cross connected and powered by battery 31 or 32 and both DC subsystems remain OPERABLE (Ref.2). Similarly, DC subsystems 33 and 34 may be cross connected and powered by battery 33 or 34. However, only one pair of subsystems at a time may be cross connected. Cross connecting DC subsystems in Modes 5 and 6 and during movement of irradiated fuel is acceptable because there is no requirement for redundancy or separation between DC busses when the plant is in this condition. Both DC subsystems in the cross connected pair remain OPERABLE even when powered by one battery because the capacity of one battery is adequate to carry the loads on both busses when the plant is in this condition.

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;

b. Required features needed to mitigate a fuel handling accident are available;

c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

(continued)

BASES

APPLICABILITY (continued)

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

ACTIONS

A.1, A.2.1, A.2.2, A.2.3 and A.2.4

If any DC electrical subsystem required by LCO 3.8.10 becomes inoperable, the remaining DC power available may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystems and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystems should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.4. Therefore, see the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

REFERENCES

1. FSAR, Chapter 14.
2. FSAR, Chapter 8.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND

This LCO delineates the limits on electrolyte temperature, level, float voltage, and specific gravity for the DC power source batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.4, "DC Sources - Operating," and LCO 3.8.5, "DC Sources - Shutdown."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 14 (Ref. 1), assume Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least one train of DC sources OPERABLE during accident conditions, in the event of:

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

b. A worst case single failure.

Battery cell parameters satisfy the Criterion 3 of 10 CFR 50.36.

LCO

Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA.

(continued)

BASES

LCO (continued)

Electrolyte limits are conservatively established, allowing continued DC electrical system function even with Category A and B limits not met.

APPLICABILITY

The battery cell parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery electrolyte is only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.4 and LCO 3.8.5.

ACTIONS

The ACTIONS Table is modified by a Note which indicates that separate Condition entry is allowed for each battery. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DC subsystem. Complying with the Required Actions for one inoperable DC subsystem may allow for continued operation, and subsequent inoperable DC subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

A.1, A.2 and A.3

With one or more cells in one or more batteries not within limits (i.e., Category A limits not met, Category B limits not met, or Category A and B limits not met) but within the Category C limits specified in Table 3.8.6-1 in the accompanying LCO, the battery is degraded but there is still sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of Category A or B limits not met and operation is permitted for a limited period.

The pilot cell electrolyte level and float voltage are required to be verified to meet the Category C limits within 1 hour (Required Action A.1). This check will provide a quick indication of the status of the remainder of the battery cells.

(continued)

BASES

ACTIONS

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

One hour provides time to inspect the electrolyte level and to confirm the float voltage of the pilot cells. One hour is considered a reasonable amount of time to perform the required verification.

Verification that the Category C limits are met (Required Action A.2) provides assurance that during the time needed to restore the parameters to the Category A and B limits, the battery is still capable of performing its intended function. A period of 24 hours is allowed to complete the initial verification because specific gravity measurements must be obtained for each connected cell. Taking into consideration both the time required to perform the required verification and the assurance that the battery cell parameters are not severely degraded, this time is considered reasonable. The verification is repeated at 7 day intervals until the parameters are restored to Category A or B limits. This periodic verification is more frequent than the normal Frequency of pilot cell Surveillances because of the degraded condition of the battery.

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. With the consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable prior to declaring the battery inoperable.

B.1

With one or more batteries with one or more battery cell parameters outside the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as not completing the

(continued)

BASES

ACTIONS

B.1 (continued)

Required Actions of Condition A within the required Completion Time or average electrolyte temperature of representative cells outside the limits of SR 3.8.6.3 are also cause for immediately declaring the associated DC electrical power subsystem inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 2), which recommends regular battery inspections (at least one per month) including voltage, specific gravity, and electrolyte temperature of pilot cells.

SR 3.8.6.2

The quarterly inspection of specific gravity and voltage is consistent with IEEE-450 (Ref. 2) which recommends augmentation of the battery inspections conducted in SR 3.8.6.1 at least once per quarter by checking the level, voltage and specific gravity of each cell, and the temperature of pilot cells.

Measuring and recording the amount of water added is a trending method for those cells found with electrolyte below minimum level.

SR 3.8.6.3

This Surveillance verification that the average temperature of representative cells (i.e., every fifth cell) is within specified limits, is consistent with a recommendation of IEEE-450 (Ref. 2), that states that the temperature of electrolytes in representative cells should be determined on a quarterly basis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.6.3 (continued)

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

Table 3.8.6-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose temperature, voltage, and electrolyte specific gravity approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450 (Ref. 2), with the extra ¼ inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote a to Table 3.8.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 2) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is $\geq$ 2.13 V per cell. This value is based on the recommendations of IEEE-450 (Ref. 2), which states that prolonged operation of cells < 2.13 V can reduce the life expectancy of cells.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6-1 (continued)

The Category A limit specified for specific gravity for each pilot cell is ≥ 1.205 (0.010 below the manufacturer fully charged nominal specific gravity or a battery charging current that had stabilized at a low value). This value is characteristic of a charged cell with adequate capacity. According to IEEE-450 (Ref. 2), the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings are corrected for actual electrolyte temperature as long as level is maintained within the required range. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is ≥ 1.195 (0.020 below the manufacturer fully charged, nominal specific gravity) with the average of all connected cells > 1.205 (0.010 below the manufacturer fully charged, nominal specific gravity). These values are based on manufacturer's recommendations. The minimum specific gravity value required for each cell ensures that the effects of a highly charged or newly installed cell will not mask overall degradation of the battery.

Category C defines the limits for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6-1 (continued)

When any battery parameter is outside the Category C limits, the assurance of sufficient capacity described above no longer exists, and the battery must be declared inoperable.

The Category C limits specified for electrolyte level (above the top of the plates and not overflowing) ensure that the plates suffer no physical damage and maintain adequate electron transfer capability.

The Category C limits for float voltage is based on IEEE-450 (Ref. 2), which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limit of average specific gravity ≥ 1.195 is based on manufacturer recommendations (0.020 below the manufacturer recommended fully charged, nominal specific gravity). In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

The footnotes to Table 3.8.6-1 are applicable to Category A, B and C specific gravity. Footnote (b) to Table 3.8.6-1 requires the above-mentioned correction for electrolyte temperature.

Footnote (c) to Table 3.8.6-1 allows the float charge current to be used as an alternate to specific gravity for up to 7 days following a battery recharge.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6-1 (continued)

A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref.2). Within 7 days, each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than 7 days.

REFERENCES

1. FSAR, Chapter 14.

2. IEEE-450-1995.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters – Operating

BASES

BACKGROUND

The inverters are the preferred source of power for the 120 V AC vital instrument buses because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the vital instrument buses.

There are four 120 volt AC vital instrument buses (VIBs), Nos. 31, 32, 33 and 34. The preferred power supplies to these buses are static inverters, Nos. 31, 32, 33 and 34, which are in turn supplied from separate 125 volt DC buses, Nos. 31, 32, 33 and 34. Each of the four 125 volt DC buses is powered by a battery and associated battery charger.

Inverters 31, 32, and 33 each have an associated backup 480 V/120 V constant voltage transformer (CVT). Each of these inverters has a manual bypass switch that causes the associated VIB to receive AC power from plant AC sources via the backup CVT instead of the DC powered inverter. Inverters 31, 32, and 33 will transfer to the backup power supply (i.e., the associated CVT) automatically in the event of an inverter failure. However, the backup CVTs for inverters 31, 32, and 33 are supplied from non-safety related buses that are stripped and not automatically re-connected following a safety injection (SI) signal or a loss of offsite power (LOOP). Therefore, operator action is required to re-energize VIBs 31, 32, or 33 following an SI or LOOP if the associated inverter is being bypassed or fails during the event. Additionally, the potential exists that the bus powering the backup CVT may not be available following an event.

Inverter 34 has two associated backup 480 V/120 V constant voltage transformers (CVTs). The CVTs associated with inverter 34 are powered from separate safeguards power trains using buses that are automatically re-energized following an SI or LOOP. Inverter 34 can be manually bypassed such that either of the associated CVTs can be used to power VIB 34. Inverter 34 will not automatically transfer to a backup power supply (i.e., the

(continued)

BASES

BACKGROUND (continued)

associated CVTs) in the event of an inverter failure. Manual operator action is also needed to transfer between the CVTs capable of powering VIB 34.

Using a separate battery and inverter to power each VIB ensures a continuous source of power for the instrumentation and controls of the engineered safety features (ESF) systems and the reactor protection system (RPS) during postulated events including the loss of offsite power. This is consistent with requirements described in Generic Letter 91-011 (Ref. 1). Continuity of power to the VIBs is assured because each of the four station batteries is sized to carry its expected shutdown loads for a period of 2 hours (Ref. 2). Additionally, four battery chargers have been sized to recharge these batteries while carrying the normal DC subsystem load (Ref. 2).

Note that battery charger 34 is not required by LCO 3.8.4. This is acceptable because VIB 34 can be powered by either of the two CVTs supplied by separate safeguard power trains if battery charger 34 is not available following an event. Specific details on inverters and their operating characteristics are found in the FSAR, Chapter 8 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 14 (Ref. 3), assumes Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required 120 V AC vital instrument buses OPERABLE during accident conditions in the event of:

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

b. A worst case single failure.

The 2 CVTs capable of supplying VIB 34 are needed to ensure the availability of power to VIB 34 following the depletion of battery 34. Although battery charger 34 would normally be used to supply VIB 34 via inverter 34, battery charger 34 is not safety related and may not be available after a design basis event.

Inverters are a part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36.

LCO

The inverters (and CVTs associated with VIB 34) ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Maintaining the required inverters (and CVTs associated with VIB 34) OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. The four inverters ensure an uninterruptible supply of AC electrical power to the AC vital buses even if the 480 V safety buses are de-energized.

(continued)

BASES

LCO (continued)

Operable inverters require the associated 120 V AC vital instrument bus to be powered by the inverter with output voltage and frequency within tolerances, and power input to the inverter from a 125 VDC station battery.

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

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

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

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "Inverters – Shutdown."

ACTIONS

With an inverter inoperable, its associated VIB becomes inoperable until it is re-energized from its associated backup CVT. For this reason a Note to the Actions requires entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating." This ensures that the vital bus is re-energized within 2 hours.

A.1

With one of the two CVTs capable of supplying VIB 34 not OPERABLE, VIB 34 will be powered from battery 34 via inverter 34 for a minimum of 2 hours following the initiation of any event. After battery 34 is depleted, the second CVT capable of powering VIB 34 will maintain power to VIB 34 even if non-safety related battery charger 34 is not available. A 30 day Completion Time to restore both CVTs to OPERABLE is needed because a failure of the safeguards power train supporting the remaining CVT would result in the loss of two VIBs (i.e, VIB 34 and the VIB associated with

(continued)

BASES

ACTIONS

A.1 (continued)

the failed safeguards power train) but only after the associated batteries are depleted. A 30 day Completion Time to restore both CVTs to OPERABLE is acceptable because of the low probability of an accident in conjunction with the loss of a specific safeguards power train.

B.1

With both of the CVTs capable of supplying VIB 34 not OPERABLE, VIB 34 will be powered from battery 34 via inverter 34 for a minimum of 2 hours following the initiation of any event. After battery 34 is depleted, inverter 34 may not be available to power VIB 34 because battery charger 34 is not safety related and is powered from a non-safety related bus. Therefore, at least one CVT must be restored within 7 days.

A 7 day Completion Time to restore at least one of the two CVTs to OPERABLE is needed and is acceptable because of the following: VIB 34 will be powered from battery 34 via inverter 34 for a minimum of 2 hours; non-safety related battery charger 34 may be available following an event; and, the low probability of an event during this 7 day period.

C.1 and C.2

With an inverter inoperable, its associated VIB must be powered from its associated backup CVT. However, the backup CVTs for inverters 31, 32, and 33 are supplied from non-safety related buses that are stripped and not automatically re-connected following a SI signal or a LOOP. Both backup CVTs for inverter 34 are powered from safety related buses that may be de-energized until the associated safeguards power train is energized (i.e., diesel generator starts). Therefore, a VIB powered from a backup CVT when the associated inverter is inoperable will be and could remain de-energized following a SI signal or a LOOP.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

If a VIB will be de-energized as a result of SI signal or LOOP, a loss of safety function could exist for any VIB powered function that requires power to perform the required safety function (e.g., automatic actuation of core spray, Regulatory Guide 1.97 instrumentation, etc.) if the redundant required feature is inoperable. Therefore, Required Action C.1 requires declaring required feature(s) supported by associated inverter inoperable when its required redundant feature(s) is inoperable. As specified in the associated Note, this requirement only applies to feature(s) that require power to perform the required safety function. The 2 hour Completion Time is consistent with LCO 3.8.9, AC Distribution System - Operating, requirements for an inoperable VIB.

With an inverter inoperable and its associated VIB powered from its associated backup CVT, there is increased potential for inadvertent actuation for ESFAS or RPS functions, especially if redundant channels are inoperable and in the tripped condition. This is because these de-energize to actuate functions are relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the VIBs is the preferred source for powering instrumentation trip setpoint devices. Therefore, only one inverter may be inoperable at one time and an inoperable inverter must be restored to OPERABLE within 7 days. The 7 day Completion Time is needed because it ensures that the VIBs are powered from the uninterruptible inverter source. The 7 day Completion Time is acceptable because Required Action C.1 ensures that an inoperable inverter does not result in a loss of any safety function. The 7 day Completion Time is consistent with commitments made in response to Generic Letter 91-011 (Ref. 1).

D.1 and D.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply.

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions. Frequency verification is not required for inverter 34 because there is no installed instrumentation for indicating this parameter.

SR 3.8.7.2

This Surveillance verifies that the power supply to VIB 34 can be manually transferred from the inverter to each of the required CVTs. This SR ensures that power to VIB 34 can be maintained after the depletion of battery 34. The 24 month Frequency takes into account that either of the CVTs is capable of performing this safety function and the demonstrated reliability of this equipment.

(continued)

BASES (continued)

REFERENCES

1. Generic Letter 91-011, Resolution of Generic Issues 48, "LCOs for Class 1E Vital Instrument Buses," and 49, "Interlocks and LCOS for Class 1E Tie Breakers" pursuant to 10 CFR 50.54(f).
2. FSAR, Chapter 8.
3. FSAR, Chapter 14.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters – Shutdown

BASES

BACKGROUND

A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 14 (Ref. 1), assume Engineered Safety Feature systems, including inverters that supply required 120 V AC vital instrument buses, are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protective System and Engineered Safety Features Actuation System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of one inverter to each VIB bus during MODES 5 and 6 and when moving irradiated fuel ensures that:

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

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

c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36.

LCO

The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The battery powered inverters provide uninterruptible supply of AC electrical power to the VIBs even if the 480 V safety buses are de-energized. OPERABILITY of the inverters requires that the VIB be powered by the inverter. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

This LCO does not require OPERABILITY of the constant voltage transformers (CVTs) capable of supplying VIB 34 even if inverter 34 is required to be OPERABLE. This is acceptable because VIB 34 will be powered from battery 34 via inverter 34 for a minimum of 2 hours and electrical buses may be cross connected as needed to support inverter 34 prior to the depletion of battery 34.

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

(continued)

BASES

APPLICABILITY (continued)

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

A.1, A.2.1, A.2.2, A.2.3 and A.2.4

If more than one VIB is required by LCO 3.8.10, "Distribution Systems – Shutdown," the remaining OPERABLE Inverters may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for positive reactivity additions. By the allowance of the option to declare required features inoperable with the associated inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverters and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention.

(continued)

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3 and A.2.4 (continued)

The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a constant voltage source transformer.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and VIBs energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the VIBs. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions. Frequency verification is not required for inverter 34 because there is no installed instrumentation for indicating this parameter.

REFERENCES

1. FSAR, Chapter 14.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems – Operating

BASES

BACKGROUND

The onsite AC, DC, and 120 V AC vital instrument bus VIB electrical power distribution systems are divided into three safeguards power trains (5A, 2A/3A and 6A) consisting of four 480 VAC safeguards buses and associated AC electrical power distribution subsystems, four 125 VDC bus subsystems, and four VIBs.

The safeguards subsystems are arranged in three trains such that any two trains are capable of meeting minimum requirements for accident mitigation or safe shutdown. The three safeguards subsystems consist of 480 volt bus 5A (associated with DG 33), 480 volt bus 6A (associated with DG 32), and 480 volt buses 2A and 3A (associated with DG 31). Buses 2A and 3A are considered a single safeguards bus. The electrical subsystems are identified in Table B 3.8.9-1.

The AC electrical power subsystem for each train consists of an Engineered Safety Feature (ESF) 480 V bus and motor control centers. Each 480 V bus has at least one offsite source of power as well as a dedicated onsite diesel generator (DG) source. Each of the four 480 V volt buses can receive offsite power from either the normal (138 kV) or alternate (13.8 kV) offsite source. The normal offsite power source uses either of the two 138 kilovolt (kV) ties from the Buchanan substation. The alternate offsite power source uses either of the two 13.8 kV ties from the Buchanan substation. There is no automatic transfer from the normal to the alternate source of offsite power.

Offsite power to 480 V buses 5A and 6A is supplied from 6.9 kV buses 5 and 6, respectively, which in turn receive power from either 138 kV offsite feeder via the Station Auxiliary Transformer (SAT). Alternately, 6.9 kV buses 5 and 6 can be supplied from either of the two 13.8 kV ties via an auto-transformer associated with the 13.8 kV feeder being used.

(continued)

BASES

BACKGROUND (continued)

When the plant is at power, 480 V buses 2A and 3A are normally powered from the Main Generator via the Unit Auxiliary Transformer (UAT) and the 6.9 kV buses 2 and 3 via SSTs 2 and 3. When the plant is not operating, buses 2A and 3A are supplied from 6.9 kV buses 5 and 6, respectively, via tie breakers. Following a unit trip, power to 480 V buses 2A and 3A is maintained by a fast transfer that connects buses 2A and 3A to power supplied from offsite to 6.9 kV buses 5 and 6. If the 138 kV system is not available, either of the two independent 13.8 kV feeders can be connected to the 6.9 kV buses through associated 20 MVA 13.8 KV/6.9 KV auto-transformers. When the 13.8 kV power source is used to feed 6.9 kV buses 5 and 6 and the main generator is used to feed 6.9 kV buses 1, 2, 3 and 4, automatic transfer of the 6.9 KV buses 1, 2, 3 and 4 to the 13.8 kV source following a unit trip must be prohibited to prevent overloading of the 13.8 kV auto-transformer. Therefore, a unit trip when a 13.8 kV power source is used to feed 6.9 kV buses 5 and 6 will result in 480 V busses 2A and 3A being de-energized and subsequently being powered from DG 31.

Each of the three 480 V safeguards subsystems receives DC control power from its associated battery charger and battery source. Battery No. 31 supplies DC control power to safeguards power train 5A including DG 33. Battery No. 32 supplies DC control power to safeguards power train 6A including DG 32. Battery No. 33 supplies DC control power to safeguards power train 2A/3A including DG 31. Batteries 31 and 32 also supply ESFAS and RPS trains A and B, respectively. Additional description of this system may be found in the Bases for LCO 3.8.1, "AC Sources – Operating," and the Bases for LCO 3.8.4, "DC Sources – Operating."

The AC electrical power distribution system for each train includes the safety related motor control centers shown in Table B 3.8.9-1.

There are four 120 volt vital AC instrument buses (VIBs), each consisting of two interconnected buses. The four VIBs are powered by static inverters that are powered from the four separate 125 volt DC buses.

(continued)

BASES

BACKGROUND (continued)

Inverters 31, 32, and 33 each have an associated backup 480 V/120 V constant voltage transformer (CVT). Each of these inverters has a manual bypass switch that causes the associated VIB to receive AC power from plant AC sources via the backup CVT instead of the DC powered inverter. Inverters 31, 32, and 33 will transfer to the backup power supply (i.e., the associated CVT) automatically in the event of an inverter failure. However, the backup CVTs for inverters 31, 32, and 33 are supplied from non-safety related buses that are stripped and not automatically re-connected following a safety injection (SI) signal or a loss of offsite power (LOOP). Therefore, operator action is required to re-energize VIBs 31, 32, or 33 following an SI or LOOP if the associated inverter is being bypassed or fails during the event. Additionally, the potential exists that the bus powering the backup CVT may not be available following an event.

Inverter 34 has two associated backup 480 V/120 V constant voltage transformers (CVTs). The CVTs associated with inverter 34 are powered from separate safeguards power trains using buses that are automatically re-energized following an SI or LOOP. Inverter 34 can be manually bypassed such that either of the associated CVTs can be used to power VIB 34. Inverter 34 will not automatically transfer to a backup power supply (i.e., the associated CVTs) in the event of an inverter failure. Manual operator action is also needed to transfer between the CVTs capable of powering VIB 34.

The 125 volt DC system is divided into four buses with one battery and battery charger (supplied from the 480 volt system) serving each. The battery chargers supply the normal DC loads as well as maintaining proper charges on the batteries. The DC system is redundant from battery source to actuation devices which are powered from the batteries. Four batteries feed four DC power panels, which in turn feed major loads, such as instrument bus inverters and switchgear control circuits. DC power panels 31 and 32 feed DC distribution panels, which in turn feed relaying and instrumentation loads. Continuity of power to the VIBs is assured because each of the four station batteries is sized to carry its expected shutdown loads for a period of 2 hours.

(continued)

BASES

BACKGROUND (continued)

Additionally, four battery chargers have been sized to recharge these batteries while carrying the normal DC subsystem load (Ref. 2).

Note that battery charger 34 is not required by LCO 3.8.4, DC Sources - Operating. This is acceptable because VIB 34 can be powered by either of the two CVTs supplied by separate safeguard power trains if battery charger 34 is not available following an event. The 2 CVTs capable of supplying VIB 34 are needed to ensure the availability of power to VIB 34 following the depletion of battery 34. Although battery charger 34 would normally be used to supply VIB 34 via inverter 34, battery charger 34 is not safety related and may not be available after a design basis event.

The list of all required distribution buses is presented in Table B 3.8.9-1.

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 14 (Ref. 1), assume ESF systems are OPERABLE. The AC, DC, and AC vital instrument bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC, DC, and VIB electrical power distribution systems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining power distribution systems OPERABLE during accident conditions in the event of:

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. A worst case single failure.

The distribution systems satisfy Criterion 3 of 10 CFR 50.36.

LCO

The required power distribution subsystems listed in Table B 3.8.9-1 ensure the availability of AC, DC, and VIB electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. The AC, DC, and VIB electrical power distribution subsystems are required to be OPERABLE.

Maintaining the AC, DC, and VIB electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

OPERABLE AC electrical power distribution subsystems require the associated buses and safety related motor control centers to be energized to their proper voltages. OPERABLE DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger. OPERABLE vital instrument bus electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated inverter via inverted DC voltage or constant voltage transformer.

In addition, tie breakers between redundant safety related AC, DC, and VIB power distribution subsystems must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, that could cause the failure of a redundant subsystem and a loss of essential safety function(s). If any tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety

(continued)

BASES

LCO (continued)	related redundant electrical power distribution subsystems. It does not, however, preclude redundant 480 V buses from being powered from the same offsite circuit.

APPLICABILITY

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

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

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

Electrical power distribution subsystem requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.10, "Distribution Systems – Shutdown."

ACTIONS

A.1

With one or more required AC buses or motor control centers (except VIBs) in one train inoperable, the remaining AC electrical power distribution subsystems in the other trains are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure and that redundant required features are OPERABLE. The overall reliability is reduced, however, because a single failure in the remaining power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required AC buses and motor control centers must be restored to OPERABLE status within 8 hours.

Condition A worst scenario is one train without AC power (i.e., no offsite power to the train and the associated DG inoperable). In this Condition, the unit is more vulnerable to a loss of the minimum required AC power. It is, therefore, imperative that the

(continued)

BASES

ACTIONS

A.1 (continued)

unit operator's attention be focused on minimizing the potential for loss of power to the remaining trains by stabilizing the unit, and on restoring power to the affected train. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because of:

a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train, to the actions associated with taking the unit to shutdown within this time limit; and

b. The potential for an event in conjunction with a single failure of a redundant component in the train with AC power.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 2 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the AC distribution system. At this time, a DC circuit could again become inoperable, and AC distribution restored OPERABLE. This could continue indefinitely.

The Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition A was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

(continued)

BASES

ACTIONS (continued)

B.1

With one VIB inoperable, the remaining OPERABLE AC vital instrument buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition assuming redundant required features are inoperable. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the required AC vital instrument bus must be restored to OPERABLE status within 2 hours by powering the bus from the associated inverter via inverted DC, or constant voltage transformer.

Condition B represents one VIB without power; potentially both the DC source and the associated AC source are nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of minimum required noninterruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining vital instrument buses and restoring power to the affected vital instrument bus.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate vital instrument bus AC power. Taking exception to LCO 3.0.2 for components without adequate vital instrument bus AC power, that would have the Required Action Completion Times shorter than 2 hours if declared inoperable, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate VIB AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

(continued)

BASES

ACTIONS

B.1 (continued)

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time takes into account the importance to safety of restoring the VIB to OPERABLE status, the redundant capability afforded by the other OPERABLE vital buses, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the VIB distribution system. At this time, an AC train could again become inoperable, and VIB distribution restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition B was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1

With one DC bus inoperable, the remaining DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure and that redundant required features are OPERABLE. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystems could result in the minimum required ESF functions not being supported.

(continued)

BASES

ACTIONS

C.1 (continued)

Therefore, the required DC buses must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery or charger.

Condition C represents one train without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a loss of minimum required DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining trains and restoring power to the affected train.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and

c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 2). The second Completion Time for Required Action C.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for

(continued)

BASES

ACTIONS

C.1 (continued)

instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the DC distribution system. At this time, an AC train could again become inoperable, and DC distribution restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition C was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

D.1 and D.2

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

E.1

With one or more trains with inoperable distribution subsystems that result in a loss of safety function, adequate core cooling, containment OPERABILITY and other vital functions for DBA mitigation would be compromised, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the AC, DC, and AC vital instrument bus electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained, and the appropriate voltage is available to each required bus. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the AC, DC, and AC vital instrument bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. FSAR, Chapter 14.
2. Regulatory Guide 1.93, December 1974.

Table B 3.8.9-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems

TYPE	VOLTAGE	Safeguards Power Train 5A (DG 33)	Safeguards Power Train 2A/3A (DG 31)	Safeguards Power Train 6A (DG 32)	
AC Electrical Power Distribution subsystems	480 V	bus 5A[1] MCC 36A MCC 36E	bus 2A[1] bus 3A[1] MCC 36C	bus 6A[1] MCC 36B MCC 36D	
AC vital[(4)] instrument buses (VIBs)	120 V	bus 31 bus 31A	bus 33 bus 33A	bus 32 bus 32A	bus 34[3] bus 34A[3]
DC buses	125 V	bus 31[2]	bus 33[2]	bus 32[2]	bus 34[2]

(1) Tie breakers must be open between buses 5A and 2A and between buses 3A and 6A.

(2) Tie breakers between DC buses must be open.

(3) The AC Power supply to the VIB 34 and VIB 34A is supplied from MCC 36B or MCC 36C as described in the Bases for LCO 3.8.7, Inverters - Operating.

(4) Each bus pair (e.g., 31 and 31A) constitutes a single vital instrument bus.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems – Shutdown

BASES

BACKGROUND

A description of the AC, DC, and 120 V AC vital instrument bus (VIB) electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter 14 (Ref. 1), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC, DC, and VIB electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC, DC, and VIB electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC, DC, and VIB electrical power distribution subsystems during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that:

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

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

c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The AC and DC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36.

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components – all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

(continued)

BASES

APPLICABILITY (continued)

The AC, DC, and VIB electrical power distribution subsystems requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9.

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, A.2.4 and A.2.5

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions).

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal (RHR) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR ACTIONS would not be entered. Therefore, Required Action A.2.5 is provided to direct declaring RHR inoperable, which results in taking the appropriate RHR actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention.

(continued)

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, A.2.4 and A.2.5 (continued)

The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the AC, DC, and VIB electrical power distribution subsystems are functioning properly, with all the buses energized. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. FSAR, Chapter 14.

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS) and the refueling cavity (which includes the refueling canal) during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage tank.

(continued)

BASES

BACKGROUND (continued)

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS and the refueling cavity above the COLR limit.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pit, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in Bases for LCO 3.1.1, "SHUTDOWN MARGIN (SDM)."

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36.

(continued)

BASES (continued)

LCO	The LCO requires that a minimum boron concentration be maintained in all filled portions of the RCS and the refueling cavity (which includes the refueling canal) while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.
APPLICABILITY	This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" ensures that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible.

(continued)

BASES

ACTIONS

A.3 (continued)

In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS and the refueling cavity is within the COLR limits. For sampling purposes, the refueling cavity and canal are considered a single volume. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

A minimum Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. 10 CFR 50, Appendix A.
2. FSAR, Chapter 14.

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. Two installed source range neutron flux monitors (N-31 and N-32) are part of the Nuclear Instrumentation System (NIS). Additionally, the full range Excore Neutron Flux Detection System, which was installed to satisfy Regulatory Guide 1.97 requirements, includes two channels (N-38 and N-39) capable of monitoring the source range. The full range Excore Neutron Flux Detection System provides indication of subcritical neutron flux in the Control Room using the Qualified Safety Parameters Display System (QSPDS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The NIS installed source range neutron flux monitors are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The two source range NIS detectors sense thermal neutrons in the range from 1×10^{-1} to 5×10^{4} neutrons per square cm per second. In addition to count rate indication in the Control Room, this instrumentation annunciates a local horn and an alarm and light in the Control Room if the count rate increases above a preset level.

The full range Excore Neutron Flux Detection System uses high-sensitivity fission chambers sensing thermal neutrons in the range from 10^{-2} to 10^{10} neutrons per square cm per second. In addition to count rate indication from the QSPDS, this instrumentation is capable of supplying audible indication of the count rate in the control room.

The core subcritical neutron flux is continuously monitored by two source range neutron monitors which provide warning of any approach to criticality during refueling operations to alert operators to a potential boron dilution event. The operators are alerted to significant changes in the subcritical neutron flux by either the alarm or by monitoring the audible neutron count rate.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly. The audible count rate from a source range neutron flux monitor provides prompt and definite indication of any boron dilution. The count rate increase is proportional to the subcritical multiplication factor and allows operators to promptly recognize the initiation of a boron dilution event. Prompt recognition of the initiation of a boron dilution event is consistent with the assumptions of the safety analysis and is necessary to assure sufficient time is available for isolation of the primary water makeup source before SHUTDOWN MARGIN is lost (Ref. 2).

The source range neutron flux monitors satisfy Criterion 3 of 10 CFR 50.36.

LCO

This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE, each source range monitor must provide visual indication in the Control Room. In addition, each source range channel must provide either an alarm at a preset neutron flux level or continuous audible signal in the Control Room.

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry may also be required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

B.1

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

B.2

With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

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

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1.

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. 10 CFR 50, Appendix A.

2. FSAR, Section 14.1.

B 3.9 REFUELING OPERATIONS

B 3.9.3 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 OPERABILITY." Containment closure means that all potential escape paths are closed, except for the OPERABLE Purge System Penetration. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

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, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

In lieu of maintaining the equipment hatch in place for containment closure, a temporary closure device may be used to maintain containment closure during core alterations or during

(continued)

BASES

BACKGROUND (continued)

movement of irradiated fuel assemblies within containment. The temporary closure device may provide penetrations for temporary services or personnel access. The temporary closure device will be designed to withstand a seismic event and designed to withstand a pressure which ensures containment closure during refueling operations. The closure device will provide the same level of protection as that of the equipment hatch for the fuel handling accident by restricting direct air flow from the containment to the environment.

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." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge System consists of the 36-inch containment purge supply and exhaust ducts. The supply system includes roughing filters, heating coils, fan and a containment penetration with two butterfly valves for isolation. The exhaust system includes a containment penetration with two butterfly valves for isolation and can be aligned to discharge to the

(continued)

BASES

BACKGROUND (continued)

atmosphere through the plant vent either directly or through the Containment Purge Filter System (i.e., a filter bank with roughing, HEPA and charcoal filters).

The Containment Purge System must be isolated when in Modes 1, 2, 3 or 4 in accordance with requirements established in LCO 3.6.3, Containment Isolation Valves. In Modes 5 and 6, the Containment Purge System may be used for containment ventilation. When open, the Containment Purge System isolation valves are capable of closing in response to the detection of high radiation levels in accordance with requirements established in LCO 3.3.6, Containment Purge and Pressure Relief Isolation Instrumentation (Ref. 1). Despite this isolation capability, the Containment Purge System must be aligned to discharge through the Containment Purge Filter System during CORE ALTERATIONS or movement of irradiated fuel until the reactor has been shutdown for a specified minimum number of hours.

The Containment Pressure Relief Line (i.e., Containment Vent) consists of a single 10-inch containment vent line that is used to handle normal pressure changes in the Containment when in Modes 1, 2, 3 and 4 (Ref. 1). The Containment Pressure Relief Line is equipped with three quick-closing butterfly type isolation valves, one inside and two outside the containment which isolate automatically in accordance with requirements established in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation", and LCO 3.3.6, "Containment Purge System and Pressure Relief Line Isolation Instrumentation." Although the Containment Pressure Relief Line discharges to the atmosphere via the Containment Auxiliary Charcoal Filter System (i.e., a filter bank with roughing, HEPA and charcoal filters), the Containment Pressure Relief Line must remain isolated during CORE ALTERATIONS or movement of irradiated fuel until the reactor has been shut down for a specified minimum number of hours. The Containment Pressure Relief Line must remain isolated because the Containment Auxiliary Charcoal Filter System is not required to be tested in accordance with Specification 5.5.10, Ventilation Filter Test Program.

(continued)

BASES

BACKGROUND (continued)

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved in accordance with 10 CFR 50.59 and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements.

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). Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The release of radioactivity from the containment following a fuel handling accident is limited by the following:

a) The requirements of LCO 3.9.6, "Refueling Cavity Water Level;"

b) The minimum decay time of 145 hours prior to CORE ALTERATIONS; and,

c) The requirements of this LCO to either isolate the Containment Purge System or align the system to discharge through the HEPA filters and charcoal adsorbers for a minimum of first 550 hours following the reactor shutdown.

This combination of requirements ensures 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36.

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 system penetrations. For the OPERABLE containment purge system penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge isolation instrumentation. Additionally, the requirement to isolate the Containment Purge System or align the system to discharge through the HEPA filters and charcoal adsorbers for a minimum of the first 550 hours following the reactor shutdown is required to limit offsite radiation exposure to within required limits. The Containment Pressure Relief Line must remain isolated because the Containment Auxiliary Charcoal Filter System is not required to be tested in accordance with Specification 5.5.10, Ventilation Filter Test Program. The OPERABILITY requirements for this LCO ensure that the automatic purge system valves meet the assumptions used in the safety analysis to ensure that releases through the valves are filtered and can be terminated, such that radiological doses are within the acceptance limit.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the

(continued)

BASES

APPLICABILITY (continued)

potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Purge system isolation instrumentation not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1 (continued)

verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.3.2

This SR requires periodic verification every 7 days that the Containment Building Purge System is either isolated or aligned to discharge through the HEPA filters and charcoal adsorbers. This SR is needed because it requires periodic verification that LCO 3.9.3.d is being met. A Note provides the allowance that this SR is not required to be performed or met if the reactor has been subcritical for ≥ 550 hours. These restrictions ensure that the offsite dose limit for a fuel handling accident of 75 rem to the thyroid at the exclusion area boundary (i.e., 25 percent of the 10 CFR Part 100 limit of 300 rem) is met by either filtering any release from the containment or by allowing a greater decay time before fuel handling activities are permitted.

SR 3.9.3.3

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on an actual or simulated high radiation signal. The 92 day Frequency ensures that this SR is performed prior to this function being required and periodically thereafter. In LCO 3.3.6, the Containment Purge system isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24 months a CHANNEL CALIBRATION is performed. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.3 (continued)

These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

SR 3.9.3.4

This SR verifies that the required Containment Building Purge System testing is performed in accordance with Specification 5.5.10, Ventilation Filter Test Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

REFERENCES

1. FSAR, Section 5.3.
2. FSAR, Section 14.2.
3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Residual Heat Removal (RHR) and Coolant Circulation – High Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) or regulating service water or component cooling water flow. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit securing the RHR pump for short durations, under the condition that the boron concentration is not diluted. This conditional securing of the RHR pump does not result in a challenge to the fission product barrier. The RHR System meets Criterion 4 of 10 CFR 50.36.

(continued)

BASES (continued)

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one RHR loop must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path. The flow path starts in loop 32 RCS hot leg and is returned to the RCS cold legs.

The LCO is modified by a Note that allows the required operating RHR loop to be removed from service for up to 1 hour per 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4," Reactor Coolant System (RCS)", and Section 3.5, "Emergency Core Cooling Systems (ECCS)." RHR

(continued)

BASES

APPLICABILITY (continued)

loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations.

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

(continued)

BASES

ACTIONS (continued)

A.4

If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed, using at least a single barrier, within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System.

REFERENCES

1. FSAR, Section 6.2.

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) or regulating service water or component cooling water flow. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, in order to prevent this challenge.

The RHR System meets criterion 4 of 10 CFR 50.36.

(continued)

BASES (continued)

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop consists of an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path. The flow path starts in loop 32 RCS hot leg and is returned to the RCS cold legs.

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level."

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.4, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

(continued)

BASES

ACTIONS (continued)

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3

If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed, using at least a single barrier, within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. In addition, during operation of the RHR loop with the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1 (continued)

water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room.

SR 3.9.5.2

Verification that the required pump not in operation is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. FSAR, Section 6.2.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pit. 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.

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS and movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.25 (Ref. 1). 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 all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

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 145 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

(continued)

BASES

BACKGROUND (continued)	by the water and offsite doses are maintained within allowable limits (Refs. 4 and 5). The amount of radioactivity potentially released following a fuel handling accident inside containment is further reduced by the following: a) requirements in the FSAR that delay any movement of irradiated fuel until the reactor has been subcritical for at least 145 hours; and, LCO 3.9.3, "Containment Penetrations," which requires the use of HEPA and charcoal filtration on the containment purge and pressure relief path for the first 550 hours following reactor shutdown if Vantage+ fuel is used. These additional restrictions ensure that does rates at the site boundary are well within the 10 CFR Part 100 limit of 300 rem to the thyroid following a fuel handling accident inside containment (Ref. 6). Further reductions in the amount of radioactivity potentially released following a fuel handling accident inside containment are expected because the containment will be isolated either automatically or through operator action following a fuel handling accident. Specifically, LCO 3.3.6, "Containment Purge System and Pressure Relief Line Isolation Instrumentation," requires the Operability of radiogas monitors R-11 and R-12, either of which could generate an automatic isolation signal, during the movement of irradiated fuel. Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36.
LCO	A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference 3.
APPLICABILITY	LCO 3.9.6 is applicable during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, and when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety

(continued)

BASES

APPLICABILITY (continued)

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 3.7.14, "Spent Fuel Pit Water Level."

ACTIONS

A.1 and A.2

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

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

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

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

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

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.

2. FSAR, Section 14.2.

(continued)

BASES

REFERENCES (continued)

3. NUREG-0800, Section 15.7.4.

4. 10 CFR 100.10.

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

6. Safety Evaluation Report for Amendment No. 175 to Facility Operating License No. DPR-64, July 17, 1997.