2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through correlations which have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, decided as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: uncertainties in the WRB-1 and WRB-2 correlations, plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes are considered statistically such that there is at least a, 95 percent probability with 95 percent confidence level that DNBR will not occur on the most limiting fuel rod during Condition I and II events. This establishes a design DNBR value which must be met in plant safety analyses using values of input Parameters without uncertainties.

The curves of Figure 2.1-1 shows the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^{RTP}_{\Delta H}$ and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^{N}_{\Delta H}$ at reduced power based on the expression:

$$F^{N}_{\Delta H} = F^{RTP}_{\Delta H} [1.0 + PF_{\Delta H} (1.0 - P)]$$

Where: $F^{RTP}_{\Delta H}$ is the limit at RATED THERMAL POWER in the Core Operating Limits Report (COLR).

$PF_{\Delta H}$ is the Power Factor Multiplier for $F^{N}_{\Delta H}$ specified in the COLR, and

$$P \text{ is } \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

These limiting heat flux conditions are higher than those calculated for the range of all control rod positions from FULLY WITHDRAWN to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (delta I) function of the Overtemperature trip. When the axial power

imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping and fittings are designed to ANSI B 31.1 1955 Edition while the valves are designed to ANSI B 16.5, MSS-SP-66-1964, or ASME Section III-1968, which permit maximum transient pressures of up to 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as-left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± rack calibration + comparator setting accuracy).

The Trip Setpoints used in the bistables are based on the analytical limits stated in the UFSAR. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those Reactor Protection System (RPS) channels that must function in harsh environments as defined by 10 CFR 50.49, the Trip Setpoints and Allowable Values specified in the Technical Specification Limiting Conditions for Operation (LCO's) are conservatively adjusted with respect to the analytical limits. The methodology used to calculate the Trip Setpoints is consistent with Instrument Society of America standard ISA-S67.04-1982, which is endorsed via NRC Regulatory Guide 1.105, Rev. 2. The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that the safety analyses which demonstrate that safety limits are not violated remain valid (provided the unit is operated within the LCO's at the onset of any design basis event and the equipment functions as designed).

The Trip Setpoints and Allowable Values listed in the LCO's incorporate all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 9 percent of RATED THERMAL POWER) and is auto-

matically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 9 percent of RATED THERMAL POWER).

Power Range, Neutron Flux, High Rate

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature Δ T

The Overtemperature Δ T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

Operation with a reactor coolant loop out of service below the 4 loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature delta T setpoint. Three loop operation above the 4 loop P-8 has not been evaluated and is not permitted.

Overpower Delta T

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. As a result of the new AREVA steam generators, credit is taken for the operation of this trip in the accident analyses for the protection of the reactor core following a main steam line break.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 90% of nominal full loop flow. Above 36% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below the design DNBR value during normal operational transients.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 0.9 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip set point is reached shall not exceed 0.3 seconds.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-9. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trip is an anticipatory trip which provides reactor core protection against DNB resulting from the opening of two or more pump breakers above P-7. This trip is blocked below P-7. The open/close position trip assures a reactor trip signal is generated before the low flow trip set point is reached. No credit was taken in the accident analyses for operation of this trip. The functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection system.

Specification 3.0.1 through 3.0.4 establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

Specification 3.0.1 establishes the Applicability statement within each individual specification as the requirement for when (i.e., in which OPERATIONAL MODES or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, a shutdown is required to place the facility in a MODE or condition in which the specification no longer applies. The ACTIONS for not meeting a single LCO adequately manage any increase in plant risk, provided any unusual external conditions (e.g., severe weather, offsite power instability) are considered. In addition, the increased risk associated with simultaneous removal of multiple structures, systems, trains or components from service is assessed and managed in accordance with 10 CFR 50.65(a)(4).

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered a MODE in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

Specification 3.0.2 establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirements within the specified time interval constitutes compliance with a specification and (2) completion of the remedial measures of the ACTION requirements is not required when compliance with a Limiting Condition of Operation is restored within the time interval specified in the associated ACTION requirements.

Specification 3.0.3 establishes the shutdown ACTION requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated ACTION requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown MODE when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Conditions for Operation and its ACTION requirements. Planned entry into LCO 3.0.3 should be avoided. If it is not practicable to avoid planned entry into LCO 3.0.3, plant risk should be assessed and managed in accordance with 10 CFR 50.65(a)(4), and the planned entry into LCO 3.0.3 should have less effect on plant safety than other practicable alternatives. One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to enter lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the ACTION requirements are completed, the shutdown may be terminated. The time limits of the ACTION requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the ACTION requirements have been met, the LCO is no longer applicable, or the time limits of the ACTION requirements have not expired, thus providing an allowance for the completion of the required actions.

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in the COLD SHUTDOWN MODE when a shutdown is required during the POWER MODE of operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for entering the next lower MODE of operation applies. However, if a lower MODE of operation is entered in less time than allowed, the total allowable time to enter COLD SHUTDOWN, or other applicable MODE, is not reduced. For example, if HOT STANDBY is entered in 2 hours, the time allowed to reach HOT SHUTDOWN is the next 11 hours because of the total time to reach HOT SHUTDOWN is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to POWER operation, a penalty is not incurred by having to enter a lower MODE in less than the total time allowed.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% Δk/k is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than or equal to 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% Δk/k shutdown margin provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the accident and transient analyses.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC) (Continued)

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analysis to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the FSAR analysis to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) subtracting from this value the largest differences in MTC observed between EOL, all rods withdrawn, RATED THERMAL POWER conditions, and those most adverse conditions of moderator temperature and pressure, rod insertion, axial power skewing, and xenon concentration that can occur in normal operation and lead to a significantly more negative EOL MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the FSAR analysis into the limiting End Of Cycle Life (EOL) MTC value. The 300 ppm surveillance limit MTC value represents a conservative value at a core condition of 300 ppm equilibrium boron concentration that is obtained by correcting the limiting EOL MTC for burnup and born concentration.

The surveillance requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains with its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the P-12 interlock is above its allowable setpoint, 4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 5) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, and 5) offsite power or an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature ≥ 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability (minimum boration volume) requirement is established to conservatively bound expected operating conditions throughout core operating life. The analysis assumes that the most reactive control rod is not inserted into the core. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires borated water from a boric acid tank in accordance with TS Figure 3.1-2, and additional makeup from either: (1) the second boric acid tank and/or batching, or (2) a maximum of 41,800 gallons of 2,300 ppm borated water from the refueling water storage tank. With the refueling water storage tank as the only borated water source, a maximum of 73,800 gallons of 2,300 ppm borated water is required. However, to be consistent with the ECCS requirements, the RWST is required to have a minimum contained volume of 350,000 gallons during operations in MODES 1, 2, 3 and 4.

The boric acid tanks, pumps, valves, and piping contain a boric acid solution concentration of between 3.75% and 4% by weight. To ensure that the boric acid remains in solution, the tank fluid temperature and the process pipe wall temperatures are monitored to ensure a temperature of 63°F, or above is maintained. The tank fluid and pipe wall temperatures are monitored in the main control room. A 5°F margin is provided to ensure the boron will not precipitate out.

Should ambient temperature decrease below 63°F, the boric acid tank heaters, in conjunction with boric acid pump recirculation, are capable of maintaining the boric acid in the tank and in the pump at or about 63°F. A small amount of boric acid in the flowpath between the boric acid recirculation line and the suction line to the charging pump will precipitate out, but it will not cause flow blockage even with temperatures below 50°F.

With the RCS temperature below 350°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE OPERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200 °F is sufficient to provide a SHUTDOWN MARGIN of 1% delta k/k after xenon decay and cooldown from 200 °F to 140 °F. This condition requires either 2,600 gallons of 6,560 ppm borated water from the boric acid storage tanks or 7,100 gallons of 2,300 ppm borated water from the refueling water storage tank.

The 37,000 gallons limit in the refueling water storage tank for Modes 5 and 6 is based upon 21,210 gallons that is undetectable due to lower tap location, 8,550 gallons for instrument error, 7,100 gallons required for shutdown margin, and an additional 140 gallons due to rounding up.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.0 and 10.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod mis-alignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. OPERABLE condition for the analog rod position indicators is defined as being capable of indicating rod position to within the allowed rod misalignment relative to the bank demand position for a range of positions. For the Shutdown Banks and Control Bank A this range is defined as the group demand counter indicated position between 0 and 30 steps withdrawn inclusive, and between 200 and 230 steps withdrawn inclusive. This permits the operator to verify that the control rods in these banks are either fully withdrawn or fully inserted, the normal operating modes for these banks. Knowledge of these banks' positions in these ranges satisfies all accident analysis assumptions concerning their position. The range for Control Bank B is defined as the group demand counter indicated position between 0 and 30 steps withdrawn inclusive, and between 160 and 230 steps withdrawn inclusive. For Control Banks C and D the range is defined as the group demand counter indicated position between 0 and 230 steps withdrawn. Comparison of the group demand counters to the bank insertion limits with verification of rod position with the analog rod position indicators (after thermal soak after rod motion) is sufficient verification that the control rods are above the insertion limits. The full out position will be specified in the reload analysis for the cycle. This position will be within the band established by FULLY WITHDRAWN and will be administratively controlled. This band is allowable to minimize RCCA wear, consistent with Information Notice 87-19 and RCCA examinations that were conducted during Salem Unit 2 Spring outage 2008 (2R16) by the Salem RCCA vendor AREVA NP (Refer to LAR S09-01).

The ACTION statements which permit limited variation from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. Mis-alignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation. The reactivity worth of a mis-aligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumption used in the accident analysis.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with T_{avg} >541°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified in accordance with the Surveillance Frequency Control Program with more frequent verifications required if an automatic monitoring channel is inoperable. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The terms "Shutdown Rod Position Indicator," "Analog Rod Position Indicator," "Control Rod Position Indicator," and "Rod Position Indicator" are all used in this bases section or in Technical Specifications, and all refer to indication driven by the output of the Analog Rod Position Indication (ARPI) system.

One method for determining rod position are the indicators on the control console. An alternate method of determining rod position is the plant computer. Either the control console indicator or plant computer is sufficient to comply with this specification. The plant computer receives the same input from ARPI as the control console indicators and provides resolution equivalent to or better than the control console indicators. The plant computer also provides a digital readout of rod position which eliminates interpolation and parallax errors inherent to analog scales.

Rod demand position is indicated on the control console and the plant computer. The rod demand position is a digital signal, namely a pulse, and is generated each time the Rod Control System demands a rod position step change, one pulse for each rod step. The pulses are “counted” and displayed by the control console group demand step counters. There are two group demand step counters for each bank of rods with exception of shutdown banks C and D. The plant computer also “counts” and displays the demand pulses. Only the group 1 demand position of each rod bank is displayed on the plant computer as only the group 1 pulses are routed to the plant computer. The group 1 demand position on the plant computer is, by default, called “Cont Bank A Steps” or “S/D Bank A Steps” etc. with no reference to group 1 or group 2.

As the plant computer receives the same demand pulses from the Rod Control System as the control console group demand step counters and provides equivalent resolution, the plant computer “bank step” display provides an alternate method of determining group 1 rod demand position. Either the control console group 1 demand step counter or the plant computer “bank step” display is sufficient to comply with this specification for group 1 rod demand position. Only the control console group 2 demand counter can be used to comply with the specification for group 2 rod demand.

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) meeting the DNB Design Criteria during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions with the part length control rods withdrawn from the core. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the target band in the COLR per Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits specified in the COLR while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of rated THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD are derived from the plant nuclear instrumentation system through the AFD Monitor Alarm. A control room recorder continuously displays the auctioneered high flux difference and the target band limits as a function of power level. An alarm is received any time the auctioneered high flux difference exceeds the target band limits. Time outside the target band is graphically presented on the strip chart.

Measurement of the target flux difference is accomplished by measuring the power distribution when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This measurement provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.

Alternatively, linear interpolation between the most recent measurement of the target flux differences and a predicted end of cycle value provides a reasonable update because the AFD changes due to burnup tend toward 0% AFD. When the predicted end of cycle AFD from the cycle nuclear design is different from 0%, it (the prediction) may be a better value for the interpolation.

Figure B 3/4 2-1 shows a typical monthly target band.

INFORMATION ONLY*

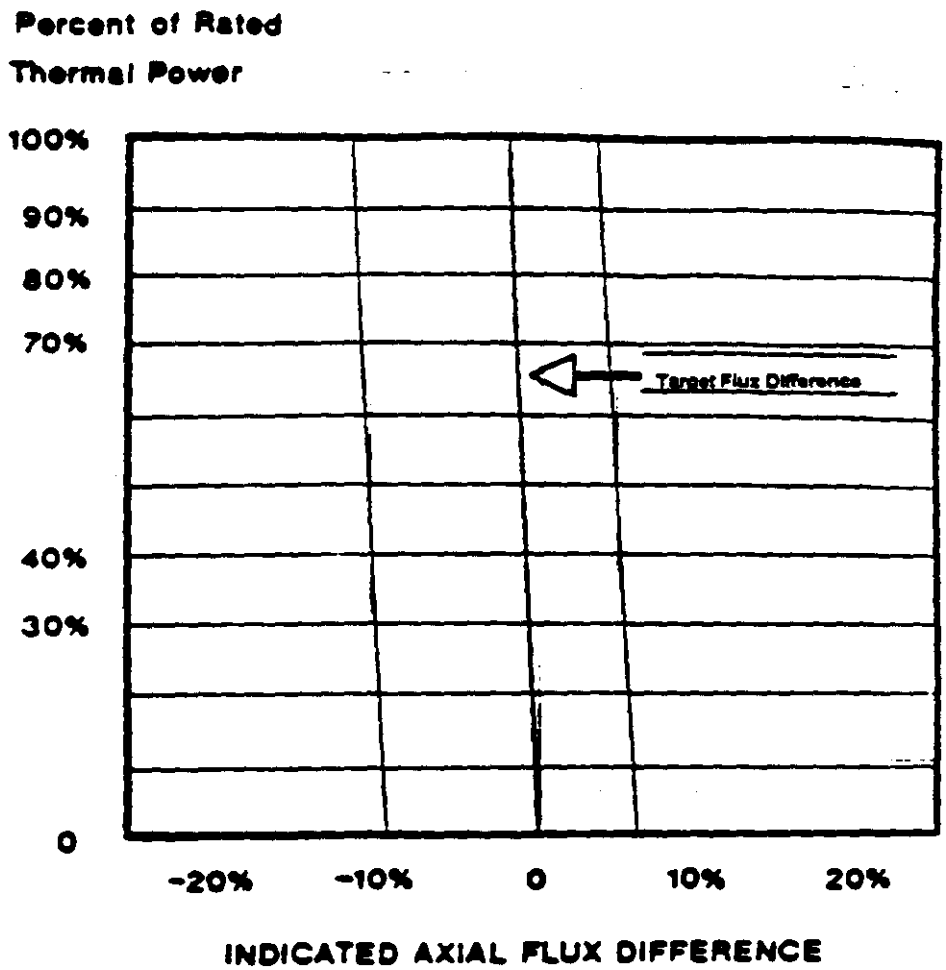


Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

* REFER TO COLR FIGURE 2 FOR ACTUAL LIMITS

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL AND RADIAL PEAKING FACTORS - $F_Q(Z)$ AND $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors and RCS flow rate ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing from the group demand position by more than the allowed rod misalignment.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.4 and 3.1.3.5 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a through d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance. For measurements obtained using the Power Distribution Monitoring System (PDMS), the appropriate measurement uncertainty is determined using the measurement uncertainty methodology contained in WCAP 12472-P-A. The cycle and plant uncertainty calculation information needed to support the PDMS calculation is contained in the COLR. The PDMS will automatically calculate and apply the correct measurement uncertainty, and apply a 3% allowance for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and is obtained from the COLR when using the PDMS or the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \leq F^{RPT}_{\Delta H}/1.08$. Where $F^{RPT}_{\Delta H}$ is the limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR). The 8% allowance is based on the following considerations:

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL AND RADIAL PEAKING FACTORS - $F_Q(Z)$ AND $F^N_{\Delta H}$ (Continued)

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q.

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics test can be compensated for in F_Q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less rapidly available.

The appropriate measurement uncertainty for $F^N_{\Delta H}$ obtained using PDMS is determined using the measurement uncertainty methodology contained in WCAP 12472-P-A. The cycle and plant specific uncertainty information needed to support the PDMS calculation is contained in the COLR. The PDMS will automatically calculate and apply the correct measurement uncertainty to the measured $F^N_{\Delta H}$.

The radial peaking factor $F_{xy}(Z)$ is measured periodically to provide assurance that the hot channel factor $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER F^{RTP}_{xy}, as provided in COLR per specification 6.9.1.9, was determined from expected power control maneuvers over the full range of burnup conditions in the core.

The core plane regions applicable to an F_{xy} evaluation exclude the following, measured in percent of core height (from the bottom of the fuel):

a. Lower core region, from 0% to 8% inclusive,
b. Upper core region, from 92% to 100% inclusive,
c. Grid plane regions at ± 2%, inclusive, and
d. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank “D” control rods.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2 hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3% from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained with the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of the design DNBR value throughout each analyzed transient.

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

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, 3) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as-left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± rack calibration + comparator setting accuracy).

The Trip Setpoints used in the bistables are based on the analytical limits stated in the UFSAR. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those Reactor Protection System (RPS) channels that must function in harsh environments as defined by 10 CFR 50.49, the Trip Setpoints and Allowable Values specified in the Technical Specification Limiting Conditions for Operation (LCO's) are conservatively adjusted with respect to the analytical limits. The methodology used to calculate the Trip Setpoints is consistent with Instrument Society of America standard ISA-S67.04-1982, which is endorsed via NRC Regulatory Guide 1.105, Rev. 2. The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that the safety analyses which demonstrate that safety limits are not violated remain valid (provided the unit is operated within the LCO's at the onset of any design basis event and the equipment functions as designed).

The Trip Setpoints and Allowable Values listed in the LCO's incorporate all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests are sufficient to demonstrate this capability. Two footnotes are added to the

CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION for Functional Units 5 and 6 of Table 4.3-1. These footnotes are consistent with Technical Specification Task Force (TSTF) Change Traveler TSTF-493, “Clarify Application of Setpoint Methodology for LSSS Functions.” The first footnote requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance, but conservative with respect to the Allowable Value. The channel evaluation verifies that channel performance continues to satisfy analysis assumptions and channel performance assumptions within the setpoint methodology. The purpose of the assessment is to ensure confidence in channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second footnote requires that the as-left setting for the channel be returned to within the as-left tolerance of the nominal Trip Setpoint. This ensures that sufficient margin is maintained to the safety limit and/or analytical limit. If the as-left channel setting cannot be returned to within the as-left tolerance of the nominal Trip Setpoint, then the channel shall be declared inoperable. The as-found tolerance for this function is calculated using the square root sum of the squares combination of uncertainty terms (rack calibration accuracy, rack measurement and test equipment accuracy, rack comparator setting accuracy, and rack drift). The as-left tolerance for this function is calculated using the square root sum of the squares combination of uncertainty terms (rack calibration accuracy, rack measurement and test equipment accuracy, and rack comparator setting accuracy). The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. Specified surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and Supplements to that report. WCAP-14333-P-A, Revision 1, “Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times,” and WCAP-15376-P-A, Revision 1, “Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times” increased the completion times and bypass test times. Out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The verification of response time provides assurance that the reactor trip and the engineered safety features actuation associated with each channel is completed within the time limit assumed in the safety analysis. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. Response time acceptance criteria have been relocated to UFSAR Section 7.2 tables and 7.3 tables. No credit is taken in the analysis for those channels with response times indicated as not applicable (i.e., N.A.). The Note 8 response times for feedwater isolation are based on WCAP-16503, “Salem Unit 1 and Unit 2 Containment Response to LOCA and MSLB for Containment Fan Cooler Unit (CFCU) Margin Recovery Project,” Revision 3, (LCR S06-10). SGFP trip and FIV closure are credited in the containment analyses for LOCA and MSLB in case an FRV fails open.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g. vendor) test

measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, “Elimination of Pressure Sensor Response Time Testing Requirements” provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types, and other components that do not have plant-specific NRC approval to use alternate means of verification, must be demonstrated by test.

The allocation for sensor response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

Channel testing in a bypassed condition shall be performed without lifting leads or jumpering bistables.

The CHANNEL CALIBRATION surveillance for the Power Range Neutron Flux Function instrumentation is modified by Note 17. Note 17 states that in MODES 1 and 2 the SSPS input relays are excluded from this surveillance when the installed bypass test capability is used to perform this surveillance. When the installed bypass test capability is used, the channel is tested in bypass versus tripped condition. To preclude placing the channel in a tripped condition, the SSPS input relays are excluded from this surveillance. The exclusion of the SSPS input relays from this test is intended to reduce the potential for an inadvertent reactor trip during surveillance testing. Therefore, the exclusion of the SSPS input relays from the surveillance is only applicable in MODES 1 and 2. The SSPS input relays must be included in the CHANNEL CALIBRATION surveillance at least once every 18 months.

The CHANNEL FUNCTIONAL TEST surveillances for the Power Range Neutron Flux and Power Range Neutron Flux High Positive Rate Function Instrumentation are modified by Note 18. Note 18 states that the SSPS input relays are excluded from this surveillance when the installed bypass test capability is used to perform this surveillance. When the installed bypass test capability is used, the channel is tested in a bypassed versus tripped condition. To preclude placing the channel in a tripped condition, the SSPS input relays are excluded from this surveillance. The exclusion of the SSPS input relays from this test is intended to reduce the potential for an inadvertent reactor trip during surveillance testing. The SSPS input relays must be included in the CHANNEL CALIBRATION surveillance at least once every 18 months.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION (Continued)

In the postulated Fuel Handling Accident, the revised dose calculations, performed using 10 CFR 50.67 and Regulatory Guide 1.183, Alternative Source Term, do not take credit for automatic containment purge isolation thus allowing for continuous monitoring of containment activity until containment closure is achieved. If required, containment purge isolation can be initiated manually from the control room.

CROSS REFERENCE - TABLE 3.3-6

T/S Table Item No.	Instrument Description	Acceptable RMS Channels
1a	Fuel Storage Area	2R5 or 2R9
1b	DELETED	
2a1a	Containment Gaseous Activity Purge & Pressure/Vacuum Relief Isolation	2R12A or 2R41A, B and D(1) (2)
2a1b	Containment Gaseous Activity RCS Leakage Detection	2R12A
2a2a	(NOT USED)	
2a2b	Containment Air Particulate Activity RCS Leakage Detection	2R11A
2b1	Noble Gas Effluent Medium Range Auxiliary Building Exhaust System (Plant Vent)	2R41B & D(1)(3)(5)
2b2	Noble Gas Effluent High Range Auxiliary Building Exhaust System (Plant Vent)	2R41C & D(1)(4)(5)
2b3	Noble Gas Effluent Condenser Exhaust System	2R15
3a	Unit 2 Control Room Intake Channel 1 (to Unit 2 Monitor) Unit 2 Control Room Intake Channel 2 (to Unit 1 Monitor) Unit 1 Control Room Intake Channel 1 (to Unit 1 Monitor) Unit 1 Control Room Intake Channel 2 (to Unit 2 Monitor)	2R1B-1 1R1B-2 1R1B-1 2R1B-2

(1) The channels listed are required to be operable to meet a single operable channel for the Technical Specification's "Minimum Channels Operable" requirement.

(2) For Modes 1, 2, 3, 4 & 5, the setpoint applies to 2R41D per Specification 3.3.3.9. The measurement range applies to 2R41A and B which display in µCi/cc using the appropriate channel conversion factor from cpm to µCi/cc.

(3) 2R41D is the setpoint channel; 2R41B is the measurement channel.

(4) 2R41D is the setpoint channel; 2R41C is the measurement channel

(5) The release rate channel 2R41D setpoint value of 2E4 uCi/sec is within the bounds of the concentration setpoint values listed in Table 3.3-6 for normal and accident plant vent flow rates.

Immediate action(s), in accordance with the LCO Action Statements, means that the required action should be pursued without delay and in a controlled manner.

3/4.3.3.2

THIS SECTION DELETED

3/4.3.3.3

THIS SECTION DELETED

3/4.3.3.4

THIS SECTION DELETED

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR 50.

3/4.3.3.6

THIS SECTION DELETED

3/4.3.3.7 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the Recommendations of Regulatory Guide 1.97, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578,"TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

The Wide Range Neutron Flux Monitors are the Gamma-Metrics Post-Accident Neutron Monitors.

3/4.3.3.8 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to UNRESTRICTED AREAS.

3/4.3.3.9

THIS SECTION DELETED

3/4.3.3.10

THIS SECTION DELETED

3/4.3.3.11

THIS SECTION DELETED

3/4.3.3.12

THIS SECTION DELETED

3/4.3.3.13

THIS SECTION DELETED

3/4.3.4 Deleted

3/4.3.3.14 POWER DISTRIBUTION MONFTORING SYSTEM (PDMS)

The Power Distribution Monitoring System (PDMS) provides core monitoring of the limiting parameters. The PDMS continuous core power distribution measurement methodology begins with the periodic generation of a highly accurate 3-D nodal simulation of the current reactor power distribution. The simulated reactor power distribution is then continuously adjusted by nodal and thermocouple calibration factors derived from an incore power distribution measurement obtained using the incore movable detectors to produce a highly accurate power distribution measurement. The nodal calibration factors are updated in accordance with the Surveillance Frequency Control Program. Between calibrations, the fidelity of the measured power distribution is maintained via adjustment to the calibrated power distribution provided by continuously input plant and core condition information. The plant and core condition data utilized by the PDMS is cross checked using redundant information to provide a robust basis for continued operation. The loop inlet temperature is generated by averaging the respective temperatures from each of the loops, excluding any bad data. The core exit thermocouples provide many readings across the core and by the nature of their usage with the PDMS, smoothing of the measured data and elimination of bad data is performed with the Surface Spline fit. PDMS uses the NIS Power Range excore detectors to provide information on the axial power distribution. Hence, the PDMS averages the data from the four Power Range excore detectors and eliminates any bad excore detector data.

The bases for the operability requirements of the PDMS is to provide assurance of the accuracy and reliability of the core parameters measured and calculated by the PDMS core power distribution monitor function. These requirements fall under four categories:

1. Assure an adequate number of operable critical sensors.

2. Assure sufficiently accurate calibration of these sensors.

3. Assure an adequate calibration database regarding the number of data sets.

4. Assure the overall accuracy of the calibration.

The minimum number of required plant and core condition inputs include the following:

1. Control Bank Positions.

2. At least 50% of the cold leg temperatures.

3. At least 75% of the signals from the power range excore detector channels (comprised of top and bottom detector section).

4. Reactor Power Level.

5. A minimum number and distribution of operable core exit thermocouples.

6. A minimum number and distribution of measured fuel assembly power distribution information obtained using the incore movable detectors is incorporated in the nodal model calibration information.

The sensor calibration of Items 1, 2, 3, and 4 above are covered under other specifications. Calibration of the core exit thermocouples is accomplished in two parts. The first being a sensor specific correction to K-type thermocouple temperature indications based on data from a cross calibration of the thermocouple temperature indications to the average RCS temperature measured via the RTDs under isothermal RCS conditions. The second part of the thermocouple calibration is the generation of thermocouple flow mixing factors that cause the radial power distribution measured via the thermocouples to agree with the radial power distribution from a full core flux map measured using the incore movable detectors. This calibration is updated in accordance with the Surveillance Frequency Control Program.

The operability requirements previously contained in Specification 3.3.3.2 have been moved to UFSAR Section 7.7.2.8 as part of Amendment 265.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, meet the DNB design criteria during all normal operations and anticipated transients. In MODES 1 and 2 with less than all coolant loops in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal for removing decay heat; but, single failure considerations require all loops be in operation whenever the rod control system is energized and at least one loop be in operation when the rod control system is deenergized.

In MODE 4, a single reactor coolant loop or RHR loop provides sufficient heat removal for removing decay heat; but, single failure considerations require that at least 2 loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires that two RHR loops be OPERABLE.

In MODE 5, single failure considerations require that two RHR loops be OPERABLE. For support systems: Service Water (SW) and Component Cooling (CC), component redundancy is necessary to ensure no single active component failure will cause the loss of Decay Heat Removal. One piping path of SW and CC is adequate when it supports both RHR loops. The support systems needed before entering into the desired configuration (e.g., one service water loop out for maintenance in Modes 5 and 6) are controlled by procedures, and include the following:

- A requirement that two RHR, two CC and two SW pumps, powered from two different vital buses be kept operable
- A listing of the active (air/motor operated) valves in the affected flow path to be locked open or disabled

Note that four filled reactor coolant loops, with at least two steam generators with at least their secondary side water level greater than or equal to 5% (narrow range), may be substituted for one residual heat removal loop. This ensures that a single failure does not cause a loss of decay heat removal.

The operation of one Reactor Coolant Pump or one RHR Pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during Boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with Boron concentration reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to 312°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer (thereby providing a volume into which the primary coolant can expand, or (2) by restricting the starting of Reactor Coolant Pumps to those times when secondary water temperature in each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 pounds per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperature. While in Mode 5 the safety valve requirement may be met by establishing a vent path of equivalent relieving capacity when no code safety valves are OPERABLE.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip setpoint is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

Surveillance testing allows a ± 3% lift setpoint tolerance. However, to allow for drift during subsequent operation, the valves must be reset to within ± 1% of the lift setpoint following testing.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control RCS pressure and establish natural circulation.

3/4.4.5 RELIEF VALVES

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

A. Manual control of PORVs to control reactor coolant system pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown.

3/4.4.5 RELIEF VALVES (continued)

B. Automatic control of PORVs to control reactor coolant system pressure. This is a function that reduces challenges to the code safety valves for overpressurization events, including an inadvertent actuation of the Safety Injection System.

C. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

D. Manual control of the block valve to : (1) unblock an isolated PORV to allow it to be used for manual and automatic control of Reactor Coolant System pressure (Items A & B), and (2) isolate a PORV with excessive seat leakage (Item C).

E. Manual control of a block valve to isolate a stuck-open PORV.

3/4.4.6 STEAM GENERATOR (SG) TUBE INTEGRITY

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

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

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

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.i, “Steam Generator (SG) Program,” and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

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

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that significantly affect burst or collapse. In that context, the term "significant" is defined as, "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

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

The accident induced leakage performance criterion ensures that the primary-to-secondary leakage caused by a design basis accident, other than a steam generator tube rupture (SGTR), is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm per SG. The accident induced leakage rate includes any primary-to-secondary leakage existing prior to the accident in addition to primary-to-secondary leakage induced during the accident.

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.7.2, "Operational Leakage," and limits primary-to-secondary leakage through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

The ACTION requirements are modified by a Note clarifying that the Actions may be entered independently for each SG tube. This is acceptable because the Action requirements provide appropriate compensatory actions for each affected SG tube. Complying with the Action requirements may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Action requirements.

If it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube plugging criteria but were not plugged in accordance with the Steam Generator Program, an evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG plugging criteria define limits on SG tube that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. An action time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity. If the evaluation determines that the affected tube(s) have tube integrity, plant operation is allowed to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering HOT SHUTDOWN following the next refueling outage or SG inspection. This allowed outage time is acceptable since operation until the next inspection is supported by the operational assessment.

If SG tube integrity is not being maintained or the Action requirements are not met, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within 36 hours. The action times are reasonable based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

During shutdown periods the SGs are inspected as required by surveillance requirements and the Steam Generator Program. NEI 97-06, “Steam Generator Program Guidelines,” and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the “as found” condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period. The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube plugging criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities and inspection locations. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines.

The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.8.4.i contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 6.8.4.i until subsequent inspections support extending the inspection interval.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. The tube plugging criteria delineated in Specification 6.8.4.i are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in size measurement and future growth. In addition, the tube plugging criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). NEI 97-06 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria. The Frequency of prior to entering HOT SHUTDOWN following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging criteria are plugged prior to subjecting the SG tubes to significant primary-to-secondary pressure differential.

3/4.4.7 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.7.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.7.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Primary to Secondary Leakage Through Any One SG

The primary-to-secondary leakage rate limit applies to leakage through any one Steam Generator. The limit of 150 gallons per day per steam generator is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines. The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, “The RCS operational primary-to-secondary leakage through any one SG shall be limited to 150 gallons per day.” The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures. The dosage contribution from the tube leakage will be within 10 CFR 50.67 limits in the event of either a steam generator tube rupture or steam line break. The analyses are based on the total primary to secondary leakage from all SGs of 1 gallon per minute as a result of accident induced conditions.

Reactor Coolant System (RCS) Pressure Isolation Valves (PIV)

The function of the RCS PIVs is to separate the high pressure RCS from the attached low pressure systems. The PIV leakage limit applies to each individual valve listed in the Technical Requirements Manual. Leakage through both series PIVs in a line must be included as part of the IDENTIFIED LEAKAGE, governed by LCO 3.4.7.2, "Operational Leakage." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 4.4.7.2.1.d). A known component of the IDENTIFIED LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not

3/4.4.7.2 OPERATIONAL LEAKAGE (Continued)

RCS operational leakage if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

Actions

Unidentified leakage or identified leakage in excess of the LCO limits must be reduced to within limits within 4 hours. This action time allows time to verify leakage rates and either identify unidentified leakage or reduce leakage to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the reactor coolant pressure boundary (RCPB). If any pressure boundary leakage exists, or primary-to-secondary leakage is not within limit, or if unidentified or identified leakage cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. The reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within 36 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary. The action 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. In COLD SHUTDOWN, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

Surveillances

Verifying RCS leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained. Pressure boundary leakage would at first appear as unidentified leakage and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. Unidentified leakage and identified leakage are determined by performance of an RCS water inventory balance. The RCS water inventory must be met with the reactor at steady state operating conditions. The surveillance is modified by a Note that the surveillance is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

3/4.4.7.2 OPERATIONAL LEAKAGE (Continued)

Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If SR 4.4.7.2.1.c is not met, compliance with LCO 3.4.6, “Steam Generator Tube Integrity,” should be evaluated. The 150 gallons per day limit is measured at room temperature (in accordance with EPRI PWR Primary-to-Secondary Leak Guidelines). If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one Steam Generator. The Surveillance is modified by a Note that states that the surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary-to-secondary leakage determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling (in accordance with EPRI PWR Primary-to-Secondary Leak Guidelines).

3/4.4.8

THIS SECTION DELETED

3/4.4.9 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Salem site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture occur since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

LCO 3.0.4.c is applicable. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS.

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G.

1) The reactor coolant temperature and pressure and system heatup and cooldown rate (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon.

   a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b) Figures 3.4-2 and 3.4-3 define limits to assure prevention of nonductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2) These limit lines shall be calculated periodically using methods provided below.

3) The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1996 Summer Addenda to Section XI of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-14040-NP-A, Rev. 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", January 1996, and ASME Boiler and Pressure Vessel Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", approved March 1999.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 32 effective full power years of service life. The 32 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. An adjusted reference temperature, (ART), based upon the fluence and the copper and nickel content of the material in question, can be predicted.

The ART is based upon the largest value of RT_{NDT} computed by the methodology presented in Regulatory Guide 1.99, Revision 2. The ART for each material is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

Initial RT_{NDT} is the reference temperature for the unirradiated material. ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by the irradiation and is calculated as follows:

$$\Delta RT_{NDT} = \text{Chemistry Factor} \times \text{Fluence Factor}$$

The Chemistry Factor, CF (F), is a function of copper and nickel content. It is given in Table B3/4.4-2 for welds and in Table B3/4.4-3 for base metal (plates and forgings). Linear interpolation is permitted.

The predicted neutron fluence as a function of Effective Full Power Years (EFPY) has been calculated and is shown in Figure B3/4.4-1. The fluence factor can be calculated by using Figure B3/4.4-2. Also, the neutron fluence at any depth in the vessel wall is determined as follows:

$$f = (f \text{ surface}) \times (e^{-0.24X})$$

where "f surface" is from Figure B3/4.4-1, and X (in inches) is the depth into the vessel wall.

Finally, the "Margin" is the quantity in °F that is to be added to obtain conservative, upper-bound values of adjusted reference temperature for the calculations required by Appendix G to 10 CFR 50.

$$\text{Margin} = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$$

If a measured value of initial RT_{NDT} for the material in question is used, σ_I may be taken as zero. If generic value of initial RT_{NDT} is used, σ_I should be obtained from the same set of data. The standard deviations, for ΔRT_{NDT}, σ_Δ, are 28°F for welds and 17°F for base metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT} surface.

The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 32 EFPY.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section XI of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50 and these methods are discussed in detail in WCAP-14040-NP-A, Rev. 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", January 1996, and ASME Boiler and Pressure Vessel Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", approved March 1999.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section XI as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation induced shift, ΔRT_{NDT} corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IC}, for the metal temperature at that time. K_{IC} is obtained from the reference fracture toughness curve, defined in ASME Code Case N-640. The K_{IC} curve is given by the equation:

$$K_{IC} = 33.2 + 20.734 \exp\left[0.02(T - RT_{NDT})\right] \qquad (1)$$

where K_{IC} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{IT} \leq K_{IC} \qquad (2)$$

where K_{IM} is the stress intensity factor caused by membrane (pressure) stress.

K_{IT} is the stress intensity factor caused by the thermal gradients.

K_{IC} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IC} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{IT}, for the reference flaw are computed. From Equation (2) the pressure stress intensity factors are obtained and from these the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IC} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IC} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stress at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature. Therefore, the K_{IC} for the 1/4T crack during heatup is lower than the K_{IC} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IC}s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the new 10CFR50 rule which addresses the metal temperature of the closure head flange regions is considered. This 10CFR50 rule states that the metal temperature of the closure flange regions must exceed the material RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Salem). Table B3/4.4-1 indicates that the limiting RT_{NDT} of 28°F occurs in the closure head flange of Salem Unit 2, and the minimum allowable temperature of this region is 148°F at pressures greater than 621 psig. These limits do not affect Figures 3.4-2 and 3.4-3.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two POPSs or an RCS vent opening of greater than 3.14 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 312°F. Either POPS has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of an Intermediate Head Safety Injection pump and its injection into a water solid RCS, or the start of a High Head Safety Injection pump in conjunction with a running Positive Displacement pump and its injection into a water solid RCS. The minimum electrical power sources required to assure POPS operability (based on POPS meeting the single failure criteria) consist of a normal (via offsite power) and an emergency (via batteries) power source for each train of POPS. Emergency diesel generators are not required for POPS to meet single failure criteria and therefore are not required for POPS OPERABILITY.

LCO 3.0.4.b is not applicable to an inoperable LTOP system when entering MODE 4. There is an increased risk associated with entering MODE 4 from MODE 5 with an inoperable LTOP system. The provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

TABLE 3/4.4-1
SALEM UNIT 2 REACTOR VESSEL TOUGHNESS DATA

Component	Plate No. or Weld No.	Material Type	Cu (%)	Ni (%)	T (°F)	50 ft-lb 35 - Mil Temp (°F)	RT (°F)	Average Upper Shell Energy: Normal to Principal Working Direction (ft-lb)	Average Upper Shell Energy: Principal Working Direction (ft-lb)
Closure Hd Dome	B4708	A533BCL1	0.11	0.70	-40	45*	-15*	82.5	127
Closure Hd Peel	B5007-3	A533BCL1	0.12	0.57	-20	15*	-20*	97*	149
Closure Hd Peel	B4707-1	A533BCL1	0.10	0.55	0	51*	0*	84*	129
Closure Hd Peel	B4707-3	A533BCL1	0.13	0.63	0	66*	6*	84*	129.5
Closure Hd Flng	B4702-1	A508CL2	-	0.68	28*	39*	28*	104*	160
Vessel Flange	B5001	A508CL2	-	0.70	12*	4*	12*	107*	164
Inlet Nozzle	B4703-1	A508CL2	-	0.69	60*	62*	60*	>72*	>111**
Inlet Nozzle	B4703-2	A508CL2	-	0.69	60*	25*	60*	>61*	>94**
Inlet Nozzle	B4703-3	A508CL2	-	0.68	60*	32*	60*	>71*	>109**
Inlet Nozzle	B4703-4	A508CL2	-	0.81	60*	40*	60*	80*	123.5
Outlet Nozzle	B4704-1	A508CL2	-	0.84	60*	8*	60*	82*	126
Outlet Nozzle	B4704-2	A508CL2	-	0.77	60*	20*	60*	75*	116
Outlet Nozzle	B4704-3	A508CL2	-	0.69	28*	8*	28*	82*	126
Outlet Nozzle	B4704-4	A508CL2	-	0.71	60*	40*	60*	77*	119
Upper Shell	B4711-1	A533BCL1	0.11	0.55	0*	50*	0*	87*	134
Upper Shell	B4711-2	A533BCL1	0.14	0.56	-10	60*	0*	79*	122
Upper Shell	B4711-3	A533BCL1	0.12	0.58	-10	88*	28*	69*	107
Inter. Shell	B4712-1	A533BCL1	0.13	0.56	0	<60	0	106	138
Inter. Shell	B4712-2	A533BCL1	0.12	0.62	-20	72	12	97	127.5
Inter. Shell	B4712-3	A533BCL1	0.11	0.57	-50	70	10	107	116
Lower Shell	B4713-1	A533BCL1	0.12	0.60	-10	68	8	98	127
Lower Shell	B4713-2	A533BCL1	0.12	0.57	-20	68	8	103	135.5
Lower Shell	B4713-3	A533BCL1	0.12	0.58	-10	70	10	121	135.5
Bottom Hd Peel	B4709-1	A533BCL1	0.12	0.60	-30	54*	-6*	90*	139
Bottom Hd Peel	B4709-2	A533BCL1	0.12	0.58	-20	42*	-18*	89*	137.5
Bottom Hd Peel	B4709-3	A533BCL1	0.11	0.56	-20	71*	11*	93*	143
Bottom Head	B4710	A533BCL1	0.12	0.60	-30	60*	0*	77*	118
Circum. Weld Bet Nozzle Shell & Int. Shell	8-442	-	0.28	0.74	-	-	-56***	-	-
Circum. Weld Bet Int. Shell & Lower Shell	9-442	-	0.197	0.060	-	-	-56***	99.7	-
Int. Shell Vertical Weld	2-442 [A,B,C]	-	0.219	0.735	-	-	-56***	96.2	-
Lower Shell Vertical Weld	3-442 [A,B,C]	-	0.213	0.867	-	-	-56***	114	-

* Estimated per NRC Standard Review Plan Section 5.8.2.
** 100% Shear not reached
*** Estimate per Pressurized Thermal Shock Rule, 10 CFR 50.61

TABLE B 3/4.4.2

CHEMISTRY FACTOR FOR WELDS, °F

Copper, Wt-%	Nickel, Wt-% 0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	21	26	27	27	27	27	27
0.03	22	35	41	41	41	41	41
0.04	24	43	54	54	54	54	54
0.05	26	49	67	68	68	68	68
0.06	29	52	77	82	82	82	82
0.07	32	55	85	95	95	95	95
0.08	36	58	90	106	108	108	108
0.09	40	61	94	115	122	122	122
0.10	44	65	97	122	133	135	135
0.11	49	68	101	130	144	148	148
0.12	52	72	103	135	153	161	161
0.13	58	76	106	139	162	172	176
0.14	61	79	109	142	168	182	188
0.15	66	84	112	146	175	191	200
0.16	70	88	115	149	178	199	211
0.17	75	92	119	151	184	207	221
0.18	79	95	122	154	187	214	230
0.19	83	100	126	157	191	220	238
0.20	88	104	129	160	194	223	245
0.21	92	108	133	164	197	229	252
0.22	97	112	137	167	200	232	257
0.23	101	117	140	169	203	236	263
0.24	105	121	144	173	206	239	268
0.25	110	126	148	176	209	243	272
0.26	113	130	151	180	212	246	276
0.27	119	134	155	184	216	249	280
0.28	122	138	160	187	218	251	284
0.29	128	142	164	191	222	254	287
0.30	131	146	167	194	225	257	290
0.31	136	151	172	198	228	260	293
0.32	140	155	175	202	231	263	296
0.33	144	160	180	205	234	266	299
0.34	149	164	184	209	238	269	302
0.35	153	168	187	212	241	272	305
0.36	158	172	191	216	245	275	308
0.37	162	177	196	220	248	278	311
0.38	166	182	200	223	250	281	314
0.39	171	186	203	227	254	285	317
0.40	175	189	207	231	257	288	320

TABLE B 3/4.4-3

CHEMISTRY FACTOR FOR BASE METAL, °F

Copper, Wt-%	Nickel, Wt-% 0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	20	20	20	20	20	20	20
0.03	20	20	20	20	20	20	20
0.04	22	26	26	26	26	26	26
0.05	25	31	31	31	31	31	31
0.06	28	37	37	37	37	37	37
0.07	31	43	44	44	44	44	44
0.08	34	48	51	51	51	51	51
0.09	37	53	58	58	58	58	58
0.10	41	58	65	65	67	67	67
0.11	45	62	72	74	77	77	77
0.12	49	67	79	83	86	86	86
0.13	53	71	85	91	96	96	96
0.14	57	75	91	100	105	106	106
0.15	61	80	99	110	115	117	117
0.16	65	84	104	118	123	125	125
0.17	69	88	110	127	132	135	135
0.18	73	92	115	134	141	144	144
0.19	78	97	120	142	150	154	154
0.20	82	102	125	149	159	164	165
0.21	86	107	129	155	167	172	174
0.22	91	112	134	161	176	181	184
0.23	95	117	138	167	184	190	194
0.24	100	121	143	172	191	199	204
0.25	104	126	148	178	199	208	214
0.26	109	130	151	180	206	216	221
0.27	114	134	155	184	211	225	230
0.28	119	138	160	187	216	233	239
0.29	124	142	164	191	221	241	248
0.30	129	146	167	194	225	249	257
0.31	134	151	172	198	228	255	266
0.32	139	155	175	202	231	260	274
0.33	144	160	180	206	234	264	282
0.34	149	164	184	209	238	268	290
0.35	153	168	187	212	241	272	298
0.36	158	173	191	216	245	275	303
0.37	162	177	196	220	248	278	308
0.38	166	182	200	223	250	281	313
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

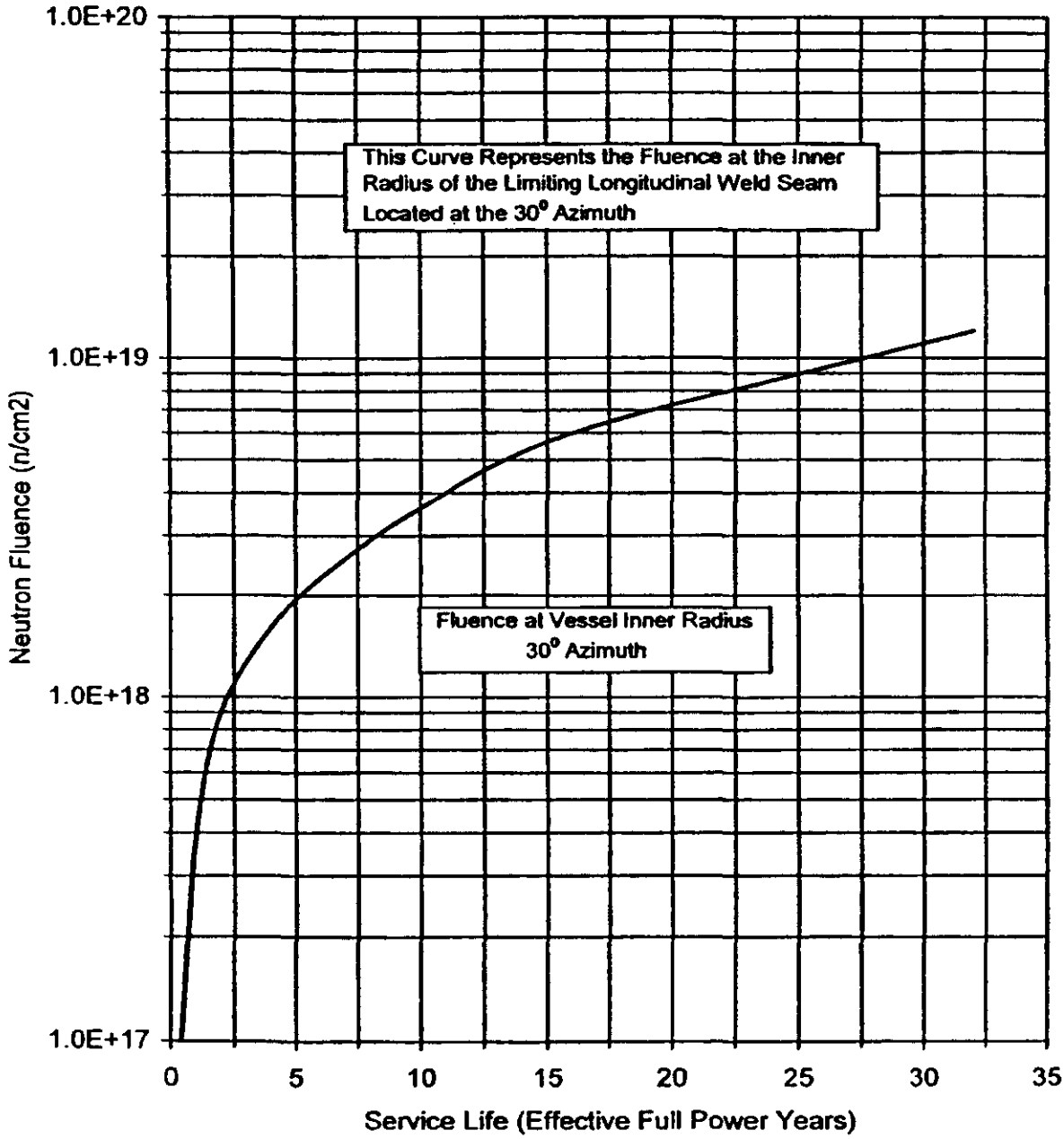
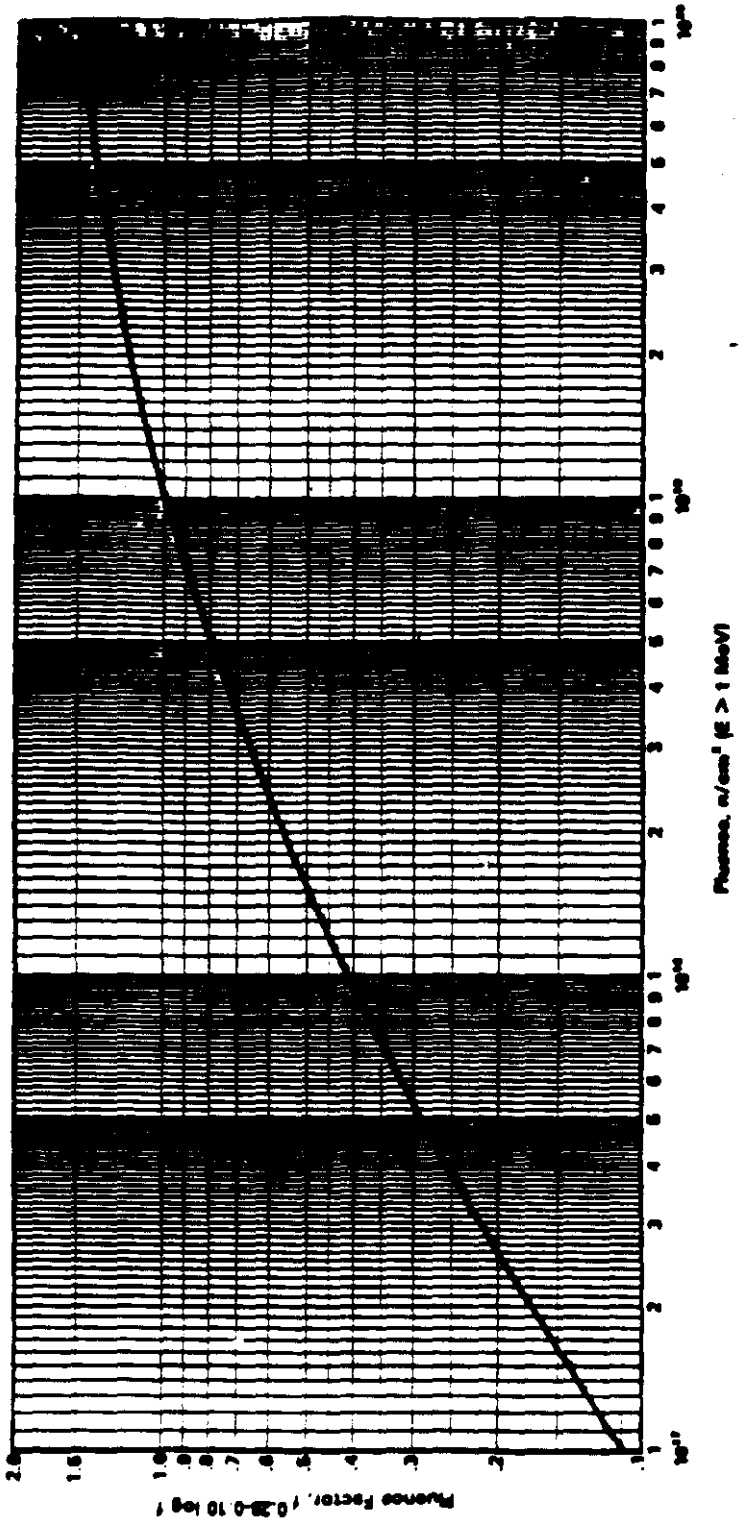


Figure B 3/4.4-1 Fast neutron fluence (E > 1 MeV) as a function of full power service life (EFPY)



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Fluence Factor for use in the expression for ΔRT_{NDT}

FIGURE B 3/4.4-2

3/4.4.11 DELETED

3/4.4.12 REACTOR VESSEL HEAD VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of a reactor vessel head vent path ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure vent in a valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System Vent Systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plant Requirements," November 1980. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each RCS accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one safety injection pump or one centrifugal charging pump to be OPERABLE and the Surveillance requirement to verify all safety injection pumps except the allowed OPERABLE safety injection pump to be inoperable below 312°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single POPS relief valve.

When running a safety injection pump with the RCS temperature less than 312 °F with the potential for injecting into the RCS and creating a mass addition pressure transient, two independent means of preventing reactor coolant system injection will be utilized. The two independent means can be satisfied by any of the following methods:
(1) A manual isolation valve locked in the closed position; or
(2) Two manual isolation valves closed; or
(3) One motor operated valve closed and its breaker de-energized and control circuit fuses removed; or
(4) One air operated valve closed and its air supply maintained in such a manner as to ensure that the valve will remain closed.

The surveillance requirements, which are provided to ensure the OPERABILITY of each component, ensure that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. The safety analyses make the assumptions with respect to: 1) both the maximum and minimum total system resistance, and 2) both the maximum and minimum branch injection line resistance. These resistances, in conjunction with the ranges of potential pump performance, are used to calculate the maximum and minimum ECCS flow assumed in the safety analyses.

The maximum and minimum flow surveillance requirements in conjunction with the maximum and minimum pump performance curves ensures that the assumptions of total system resistance and the distribution of that system resistance among the various paths are met.

The maximum total pump flow surveillance requirements ensure the pump runout limits of 560 gpm for the centrifugal charging pumps and 675 gpm for the safety injection pumps are not exceeded. Due to the effect of pump suction boost alignment, the runout limits for the surveillance criteria are ≤ 554 gpm for C/SI pumps, ≤ 664 gpm for SI pumps in cold leg alignment and ≤ 654 gpm for SI pumps in hot leg alignment.

The surveillance requirement for the maximum difference between the maximum and minimum individual injection line flows ensure that the minimum individual injection line resistance assumed for the spilling line following a LOCA is met.

LCO 3.0.4.b is not applicable to an inoperable ECCS high head subsystem when entering MODE 4. There is an increased risk associated with entering MODE 4 from MODE 5 with an inoperable ECCS high head subsystem. The provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

3/4.5.4 SEAL INJECTION FLOW

The Reactor Coolant Pump (RCP) seal injection flow restriction limits the amount of ECCS flow that would be diverted from the injection path following an ECCS actuation. This limit is based on safety analysis assumptions, since RCP seal injection flow is not isolated during Safety Injection (SI).

The LCO is not strictly a flow limit, but rather a flow limit based on a flow line resistance. Line pressure and flow must be known to establish the proper line resistance. Flow line resistance is determined by assuming that the RCS pressure is at normal operating pressure, and that the centrifugal charging pump discharge pressure is greater than or equal to 2430 psig. Charging pump header pressure is used instead of RCS pressure, since it is more representative of flow diversion during an accident. The additional LCO modifier, charging flow control valve full open, is required since the valve is designed to fail open. With the LCO specified discharge pressure and control valve position, a flow limit is established. This flow limit is used in the accident analysis.

A provision has been added to exempt surveillance requirement 4.0.4 for entry into MODE 3, since the surveillance cannot be performed in a lower mode. The exemption is permitted for up to 4 hours after the RCS pressure has stabilized within ± 20 psig of normal operating pressure. The RCS pressure requirement produces the conditions necessary to correctly set the manual throttle valves. The exemption is limited to 4 hours to ensure timely surveillance completion once the necessary conditions are established.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as a part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA.

The limits on RWST minimum volume and boron concentrations ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, (2) the reactor will remain subcritical in the cold condition following a small LOCA assuming complete mixing of the RWST, RCS, and ECCS water volumes with all control rods inserted except the most reactive control assembly (ARI-1), and (3) the reactor will remain subcritical in the cold condition following a large break LOCA (break flow area > 3.0 sq. ft.) assuming complete mixing of the RWST, RCS, and ECCS water and other sources of water that may eventually reside in the sump following a LOCA with alql control rods assumed to be out (ARO). The limits on contained water volume and boron concentration also ensure a pH value of between 7.0 and 10.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.6.1 PRIMARY CONTAINMENT

3/4 6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

The purpose of this surveillance requirement (4.6.1.1a) is not to perform any testing or valve manipulations, but to verify that containment isolation valves capable of being mispositioned are in their proper safety position (closed).

Valves and blind flanges in high radiation areas may be verified by use of administrative controls. Allowing the use of administrative means to verify compliance with the surveillance requirement for these valves is acceptable based on the limited access to these areas in Modes 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified in the proper position, is small.

Use of administrative means to verify position of valves and blind flanges that are locked, sealed or otherwise secured is acceptable for Surveillance Requirement 4.6.1.1.a. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these components, once they have been verified to be in the proper position, is low.

The service water accumulator vessel and discharge valves function to maintain water filled, subcooled fluid conditions in the containment fan coil unit (CFCU) cooling loops during accident conditions. The service water accumulator vessel and discharge valves were installed to address the Generic Letter 96-06 issues of column separation waterhammer and two phase flow during an accident involving a loss of offsite power. The operability of each service water accumulator vessel and discharge valve is required to ensure the integrity of containment penetrations associated with the containment fan coil units during accident conditions. If a service water accumulator vessel does not meet the vessel surveillance requirements, or if the discharge valve response time does not meet design acceptance criteria when tested in accordance with procedures, the containment integrity requirements of the CFCU cooling loops exclusively supplied by the inoperable accumulator vessel or discharge valve are not met. Limiting Condition for Operation 3.6.1.1 is applicable, and the cooling loops for the two CFCU's exclusively supplied by the inoperable accumulator are to be removed from service and isolated to maintain containment integrity.

3/4 6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure P_a. As an added conservatism, the measured overall integrated leakage rate (Type A test) is further limited to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the Containment Leakage Rate Testing Program.

3/4.6.1.3 CONTAINMENT AIR LOCKS

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

Each air lock is nominally a right circular cylinder, 10 feet in diameter, with a door at each end. The doors are interlocked during normal operation to prevent simultaneous opening.

During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure-seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

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

The containment air locks form part of the containment pressure boundary. 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 analysis.

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. This leakage rate is defined in 10CRF50, Appendix J as $L_a = 0.1\%$ of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure $P_a = 47.0$ psig following a DBA. The allowable leakage rate forms the basis for the acceptance criteria imposed on the surveillance requirements associated with the air locks.

Each containment air lock forms part of the containment pressure boundary. As part of the containment, the air lock safety function is related to control of the containment leakage rate resulting from 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 of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment.

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 3.9.4, “Containment Building Penetrations”.

The ACTIONS are modified by five notes. Note (1) 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. 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 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.

Note (2) adds clarification that separate condition entry is allowed for each air lock. This is acceptable, since the required ACTIONS provide appropriate compensatory measures for each inoperable air lock. Complying with the Required Actions may allow for continued operation. A subsequent inoperable air lock is governed by condition entry for that air lock.

Notes (3) and (4) ensure 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 these Notes does not affect tracking the completion time from the initial entry into condition a., only the requirement to comply with the required ACTIONS.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note (5) directs entry into the applicable Conditions and required ACTIONS of LCO 3.6.1, “Primary Containment”.

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (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. The specified time period is consistent with the ACTIONS of LCO 3.6.1.1 that requires that containment be restored to OPERABLE status within 1 hour. OPERABILITY of the air lock interlock is not required to support the OPERABILITY of an air lock door.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour completion time (ACTION a.2). 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.

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 judgement and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls.

ACTION a.3 allows the 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 to be inoperable.

Containment entry may be required on a periodic basis to perform Technical Specification Surveillances and required ACTIONS, as well as other activities on equipment inside containment that are required by Technical Specifications or activities on equipment that support Technical Specification required equipment. This Note is not intended to preclude performing other activities (i.e., non-Technical Specification required activities) if the containment is entered, using the inoperable air lock, to perform an allowed entry 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.

Because of ALARA considerations, ACTION a.3 also allows air lock doors located in high radiation areas 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.

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. In addition, ACTION b.3 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). In addition, ACTION b.3 allows air lock doors located in high radiation areas to be verified locked closed by use of administrative means.

ACTION c.1 requires that with one or more air locks inoperable for reasons other than those described in condition a. or b., action must 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 3.6.1.1) would be provided to restore the air lock door to OPERABLE status prior to requiring plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

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 3.6.1.1, 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. This completion time begins at the time that the air lock is discovered to be inoperable. 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.

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 Hot Standby within 6 hours and to Cold Shutdown within the following 30 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.

Maintaining containment airlocks OPERABLE requires compliance with the leakage rate test requirements of 10CFR50, Appendix J, as modified by approved exemptions. This Surveillance Requirement reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during 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 Appendix J, as modified by approved exemptions. Thus, the provision of Specification 4.0.2 (which allows frequency extensions) does not apply.

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

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig, and 2) the containment peak pressure does not exceed the design pressure of 47 psig during the limiting pipe break conditions. The pipe breaks considered are LOCA and steam line breaks.

The limit of 0.3 psig for initial positive containment pressure is consistent with the accident analyses initial conditions.

The maximum peak pressure expected to be obtained from a LOCA or steam line break event is ≤ 47 psig.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a LOCA or steam line break. In order to determine the containment average air temperature, an average is calculated using measurements taken at locations within containment selected to provide a representative sample of the overall containment atmosphere.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the design pressure. The visual inspections of the concrete and liner and the Type A leakage test, both in accordance with the Containment Leakage Rate Testing Program, are sufficient to demonstrate this capability.

(Note that the elements of 3/4.6.1.7 were RELOCATED to 3/4 6.3 by LCR S06-06)

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system, when operated in conjunction with the Containment Cooling System, ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

The containment spray system also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

Normal plant operation and maintenance practices are not expected to trigger surveillance requirement 4.6.2.1.d. Only an unanticipated circumstance would initiate this surveillance, such as inadvertent spray actuation, a major configuration change, or a loss of foreign material control when working within the affected boundary of the system. If an activity occurred that presents the potential of creating nozzle blockage, an evaluation would be performed by the engineering organization to determine if the amount of nozzle blockage would impact the required design capabilities of the containment spray system. If the evaluation determines that the containment spray system would continue to perform its design basis function, then performance of the air or smoke flow test would not be required. If the evaluation cannot conclusively determine the impact to the containment spray system, then the air or smoke flow test would be performed to determine if any nozzle blockage has occurred.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the spray additive system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration, ensure that 1) the iodine removal efficiency of the spray water is maintained because of the increase in pH value, and 2) corrosion effects on components within containment are minimized. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

The surveillance requirements for the service water accumulator vessels ensure each tank contains sufficient water and nitrogen to maintain water filled, subcooled fluid conditions in three containment fan coil unit (CFCU) cooling loops in response to a loss of offsite power, without injecting nitrogen covergas into the containment fan coil unit loops assuming the most limiting single failure. The surveillance requirement for the discharge valve response time test ensures that on a loss of offsite power, each discharge valve actuates to the open position in accordance with the design to allow sufficient tank discharge into CFCU piping to maintain water filled, subcooled fluid conditions in three CFCU cooling loops, assuming the most limiting single failure.

The surveillance requirements for the CFCUs ensure sufficient SWS flow through each operating cooler to provide the minimum containment cooling as assumed by the containment response analysis for a design-basis LOCA or MSLB event. The surveillance flow rate is selected to ensure adequate heat removal (with no two-phase flow). The specified surveillance flow rate represents the total flow from both the CFCU coils and the CFCU motor-cooler

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The opening of locked or sealed closed containment isolation valves (penetration flow paths) on an intermittent basis under administrative control includes the following considerations: (1) stationing a dedicated individual, who is in constant communication with the control room, at the valve controls, (2) instructing this individual to close these valves in an accident situation, and (3) assuring that the environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

The main steam isolation valves (MSIVs) fulfill their containment isolation function as remote-manual containment isolation valves. The automatic closure of the MSIVs is not required for containment isolation due to having a closed system inside containment. The remote-manual containment isolation function of the MSIVs can be accomplished through either the use of the hydraulic operator or when the MSIV has been tested in accordance with surveillance requirement 4.7.1.5 the steam assist closure function can be credited.

Surveillance Requirement (SR) 4.6.3.3 only applies to the MS7 (Main Steam Drain) valves and the MS18 (Main Steam Bypass) valves. The MS167 (Main Steam Isolation) valves are tested for main steam isolation purposes by SR 4.7.1.5. For containment isolation purposes, the MS167s are tested as remote/manual valves pursuant to Specification 4.0.5.

The containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA. Maintaining these valves (or equivalent isolation device) closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system.

A containment purge valve is not a required containment isolation valve when its flow path is isolated with a blind flange tested in accordance with SR 4.6.1.2.b. The inboard valve of both the containment purge supply and exhaust penetrations has been replaced with a testable double o-ring blind flange. The blind flange serves as the containment boundary and performs the containment integrity function in Modes 1, 2, 3, and 4. The outboard valve of both the

containment purge supply and exhaust penetrations performs no containment integrity function in MODES 1-4; these valves operate during shutdown for normal system purging and containment closure when the blind flanges are removed.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The MSSVs also provide protection against overpressurization of the Reactor Coolant Pressure Boundary by providing a heat sink for the removal of energy from the Reactor Coolant System if the preferred heat sink is not available. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 16.66 x 10^6 lbs/hr which is approximately 110% of the maximum calculated steam flow of 15.12 x 10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per OPERABLE steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with inoperable safety valves within the limitations of the ACTION requirements on the basis of the reduction in secondary steam flow associated with the required reduction of RATED THERMAL POWER. The acceptable power level (in percent RATED THERMAL POWER) for operation with inoperable safety valves was determined by performing explicit transient analysis.

The events that challenge the relief capacity of the safety valves are those resulting in decreased heat removal capability. In this category of events, a loss of external electrical load and/or turbine trip is the limiting anticipated operational occurrence. A series of cases was analyzed for this transient covering up to two inoperable safety valves on each steam generator. The results of these cases were used to determine a maximum thermal power level from which the event could be initiated without exceeding the primary and secondary side design pressure limits. Thus, the maximum allowed power level as a function of the number of inoperable MSSVs on any steam generator is presented in Table 3.7-1. Note that the power level values presented on this table are the direct inputs into the transient analysis cases and do not include any allowance for calorimetric error. Actual power level reductions must include calorimetric uncertainty and other allowances for operating margin as deemed necessary.

Specific accident analyses for RCCA Bank Withdrawal at Power scenarios demonstrate that adequate safety valve relief capacity exist with up to two inoperable safety relief valves on each steam generator. These cases demonstrate that the reactor trip on OTDT along with the relief from the available main steam safety valves is sufficient to meet secondary side pressurization limits.

For three inoperable main steam safety valves in one or more steam generators, thermal reactor power must be reduced in conjunction with a reduction in the Power Range Neutron Flux High trip setpoint to prevent overpressurization of the main steam system.

The transient analysis assumes that the MSSVs will start to open at the lift setpoint with 3% allowance for setpoint tolerance. In addition, the analysis accounts for accumulation by including a 5 psi ramp for the valve to reach its fully open position. Inoperable MSSVs are assumed to be those with the lowest lift setting. Surveillance testing as covered in Table 3.7-4 allows a ± 3% lift setpoint tolerance. However, to allow for drift during subsequent operation, the valves must be reset to within ± 1% of the lift setpoint following testing.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of offsite power.

Verifying that each Auxiliary Feedwater (AFW) pump's developed head at the flow test point is greater than or equal to the required minimum developed head ensures that the AFW pump performance has not degraded during the cycle, and that the assumption made in the accident analysis remain valid. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code. Because it is undesirable to introduce cold AFW into the steam generators while operating, the test is performed on recirculation flow. This test confirms one point on the pump design curve (head vs flow curve), and is indicative of pump performance. Inservice testing confirms pump operability, trends performance and detects incipient failures by indication of pump performance.

The flow path to each steam generator is ensured by maintaining all manual maintenance valves locked open. A spool piece consisting of a length of pipe may be used as an equivalent to a locked open manual valve. The manual valves in the flow path are: 2AF1, 21AF3, 22AF3, 23AF3, 21AF10, 22AF10, 23AF10, 24AF10, 21AF20, 22AF20, 23AF20, 24AF20, 21AF22, 22AF22, 23AF22, 24AF22, 21AF86, 22AF86, 23AF86, and 24AF86.

LCO 3.0.4.b is not applicable to an inoperable AFW train. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an AFW train inoperable. The provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

3/4.7.1.3 AUXILIARY FEED STORAGE TANK

The OPERABILITY of the auxiliary feed storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to the atmosphere concurrent with total loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the main steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

If the closure time of the main steam isolation valve (MSIV) during technical specification surveillance testing (performed at a Steam Generator pressure between 800 psig and 1015 psig) is 5.0 seconds or less and the engineered safety feature response time (including valve closure time) for the steam line isolation (MSI) signal (Table 3.3-5) is 5.5 seconds or less, then assurance is provided that MSI occurs within 12 seconds under accident conditions, where Steam Generator pressure may be lower. This method of testing assures that for main steam line ruptures that are initiated from Modes 1-3 conditions that generate a MSI signal via automatic or manual initiation and have adequate steam line pressure to close, the main steam lines isolate within the time required by the accident analysis. Fast closure of the MSIVs is assured at a minimum steam pressure of 170 psia. However, the MSIV will still close via the steam assist function between 118 – 170 psia with slightly greater closure times. For main steam line ruptures that receive an automatic or manual signal for MSI and do not have adequate steam pressure to close the MSIVs (less than 118 psia), the event does not require MSIV closure to provide protection to satisfy design basis requirements (e.g., minimum DNBR remains above the minimum DNBR limit value and peak containment pressure remains below 47 psig).

Testing for SR 4.7.1.5 is performed prior to opening the MSIVs for power operation. During testing, only one valve is opened at a time, with the other three valves remaining closed in the safe position, ensuring isolation capability is maintained. In the event of a steam line rupture, a postulated failure of the tested valve in the open position would result in the blowdown of a single steam generator since the remaining three MSIVs are closed. Failure of a single MSIV to close is consistent with the accident analysis assumptions for a major secondary system pipe rupture (UFSAR Section 15.4.2).

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on average steam generator impact values taken at 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The component cooling water system (CCW) consists of two safeguards mechanical trains supplied by three pumps powered from separate vital buses. This complement of equipment assures adequate redundancy in the event of a single active component failure during the injection phase. OPERABILITY of the CCW system exists when both loops are OPERABLE. An OPERABLE CCW loop consists of one mechanical train and one CCW pump.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.5 FLOOD PROTECTION

The limitation on flood protection ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions. The limit of elevation 10.5' Mean Sea Level is based on the elevation above which facility flood control measures are required to provide protection to safety-related equipment.

3/4.7.6 CONTROL ROOM EMERGENCY AIR CONDITIONING SYSTEM

BACKGROUND:

The control room emergency air conditioning system (CREACS) provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke.

The OPERABILITY of the CREACS ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions.

The CREACS consists of two independent, redundant trains, one from each unit that re-circulate and filter the air in the Control Room Envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air. Each CREACS train consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and fans. Ductwork, valves or dampers, doors, barriers, and instrumentation also form part of the system. The CREACS is a shared system between Unit 1 and 2 supplying a common CRE. During emergency operation following receipt of a Safety Injection or High Radiation actuation signal, for areas inside the CRE, one 100% capacity fan in each Unit's CREACS will operate in a pressurization mode with a constant amount of outside air supplied for continued CRE pressurization. One fan from each train will automatically start upon receipt of an initiation signal, with one fan in each train in standby. A failure of one fan will result in the standby fan automatically starting.

Each CREACS train has two 100% capacity fans, such that any one of the four fans is sized to provide the required flow for CRE pressurization within the common CRE during an emergency.

A failure of one CREACS filtration train requires manual actions to properly reposition dampers in support of single filtration train operation.

To minimize control room radiological doses, the CREACS outside air is supplied from the non-accident unit's emergency air intake through the cross-connected supply duct (as determined by which unit received an accident signal). Outside air is mixed with recirculated air, passed through each CREACS filter bank (pre-filter, HEPA filter, and charcoal adsorber) and cooling coil, and distributed to the common CRE. The CREACS is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding 5 Rem total effective dose equivalent (TEDE).

The CREACS is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation. Upon receipt of the actuating signal(s), normal air supply to the CRE is isolated, and the stream of ventilation air is recirculated through the system filter trains. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary. CREACS will be manually initiated in the recirculation mode only in the event of a fire outside the CRE, a toxic chemical release, or testing.

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room and other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

APPLICABLE SAFETY ANALYSES

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

The CREACS provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release, as described in UFSAR, Section 6.4. The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels, as described in UFSAR, Section 9.5.

LCO

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

In order for the CREACS trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls are proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition, when a need for CRE isolation is indicated.

A significant contributor to this system's OPERABILITY are the dampers, which are required to actuate to their correct positions. The following dampers are associated with the respective LCO*:

a.1 Fan outlet dampers: 1(2)CAA15 and 1(2)CAA16

These dampers ensure that the flow path for CREACS is operable and are required to open upon CREACS initiation. The associated fan outlet damper will open on fan operation.

a.4 Return air isolation damper: 1(2)CAA17

When aligned for single train operation, the associated air return isolation damper will be administratively controlled in the open position.

b. Other dampers required for automatic operation in the pressurization or recirculation modes:

Control Area Air Conditioning System (CAACS) outside air intake isolation dampers: 1(2)CAA40, 1(2)CAA41, 1(2)CAA43 and 1(2)CAA45

The normally open outside air intake dampers 1(2)CAA40 and inlet plenum isolation dampers 1(2)CAA43 will be closed under emergency conditions. The normally closed outside air intake dampers 1(2)CAA41 and inlet plenum isolation dampers 1(2)CAA45 are normally closed and remain closed under emergency conditions.

* Operability of the CREACS requires that each of the Unit 1 dampers are also operable

Control Area Air Conditioning System (CAACS) exhaust isolation dampers: 1(2)CAA18 and 1(2)CAA19.

These dampers are normally closed and are required to remain closed to prevent inleakage from the outside environment in the event of a toxic release.

Control Room Emergency Air Conditioning System (CREACS) air intake dampers: 1(2)CAA48, 1(2)CAA49, 1(2)CAA50 and 1(2)CAA51

CREACS outside air intake dampers are maintained closed during normal and recirculation operation and are opened automatically upon initiation of CREACS pressurization. The control logic will automatically open the CREACS air intake dampers farthest from the radiation source based upon which Unit's Solid State Protection System (SSPS) or Radiation Monitoring System (RMS) signal is received.

CAACS and CREACS interface isolation dampers: 1(2)CAA14 and 1(2)CAA20

These two dampers are normally open and do not have associated redundant dampers. These dampers serve a boundary function by isolating the CREACS from the CAACS during emergency operation of the CREACS.

Note: Dampers 1(2)CAA5, CAACS recirculation damper will receive an accident alignment signal to ensure proper accident configuration of CAACS. This damper, however, is not required for the OPERABILITY of CREACS as defined in the LCO.

APPLICABILITY

In all MODES and during movement of irradiated fuel assemblies, the CREACS must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA.

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

ACTIONS

When one CREACS train is inoperable, for reasons other than an inoperable CRE boundary, action must be taken to align CREACS for single filtration train operation within 4 hours, and restore the inoperable filtration train to OPERABLE status within 30 days. Single filtration train alignment is only permitted if the Unit with the operable CREACS train is also in Chilled Water LCO 3.7.10.a configuration. Single filtration train alignment is not permitted if in the LCO 3.7.10.c configuration. This ensures required cooling coil heat removal capacity is available. In this Condition, the remaining OPERABLE CREACS train is adequate to perform the CRE occupant protection function. With CREACS aligned for single filtration train operation and with one of the two remaining fans or associated outlet damper inoperable, restore the inoperable fan or damper to OPERABLE status within 72 hours. However, the overall reliability is reduced because a failure in the OPERABLE CREACS train could result in loss of CREACS function. The 72 hours completion time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train components to provide the required capability.

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem TEDE), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24-hour completion time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day completion time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day completion time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

In MODE 1, 2, 3, or 4, if the inoperable CREACS train or the CRE boundary cannot be restored to OPERABLE status within the required completion time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within the following 30 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 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CREACS train cannot be restored to OPERABLE status, align CREACS for single filtration train operation within 4 hours, or suspend movement of irradiated fuel assemblies. With CREACS aligned for single filtration train operation with one of the two remaining fans or associated outlet damper inoperable, restore the fan or damper to OPERABLE status within 72 hours. The 72 hours completion time is based on the ability of the remaining train components to provide the required capability.

In MODE 5 or 6, or during the movement of irradiated fuel assemblies, with two CREACS trains inoperable or with one or more CREACS trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

Immediate action(s), in accordance with the LCO Action Statements, means that the required action should be pursued without delay and in a controlled manner.

SURVEILLANCE REQUIREMENTS

Standby systems should be checked periodically to ensure that they function properly. TS Surveillance Requirement verifies that each fan is capable of operating for at least 15 minutes by initiating flow through the HEPA filter and charcoal adsorbers train(s) to ensure that the system is available in a standby mode. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

Filter testing verifies that the required CREACS testing is performed in accordance with the surveillance requirements. The surveillance requirements include 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 surveillance requirements. Filter testing will be in accordance with the applicable sections of ANSI N510 (1975) with the exception that laboratory testing of activated carbon will be in accordance with ASTM D3803 (1989). The acceptance criteria for the laboratory testing of the carbon adsorber is determined by applying a minimum safety factor of 2 to the charcoal adsorber removal efficiency credited in the design basis dose analysis as specified in Generic Letter 99-02.

Actuation testing verifies that each CREACS train starts and operates on an actual or simulated actuation signal. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

The control room envelope is considered intact and able to support operation of the CREACS when the emergency air conditioning system is capable of maintaining positive pressure with the control room boundary door(s) closed. Unfiltered air inleakage testing verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

Each CAACS normal air intake ductwork has two radiation detector channels. The two detector channels from Unit 1 and Unit 2 CAACS air intake provide input to common radiation monitor processors. Each radiation monitor processor (one for 1R1B-1/1R1B-2 and one for 2R1B-1/2R1B-2) provides a signal to initiate CREACS in the pressurization mode should high radiation be detected. A minimum of one out of two detectors in either intake will initiate the pressurization mode. With two detector channels inoperable on a Unit, operation may continue as long as CREACS is placed in-service in the pressurization or recirculation mode. Pressurization mode will be initiated after 7 days with one inoperable detector. Radiological releases during a fuel handling accident while operating in the recirculation mode could result in unacceptable radiation levels in the CRE since the automatic initiation capability has been defeated for high radiation due to isolation of the detectors. Therefore, movement of irradiated fuel assemblies or Core Alterations at either Unit will not be permitted when in the recirculation mode.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from hazardous chemicals and smoke. The testing verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, CRE boundary is inoperable. Required action allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F. These compensatory measures may also be used as required mitigating actions. Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

3/4.7.7 AUXILIARY BUILDING EXHAUST AIR FILTRATION SYSTEM

The Auxiliary Building Ventilation System (ABVS) consists of two major subsystems. They are designed to control Auxiliary Building temperature during normal and emergency modes of operation, and to contain Auxiliary Building airborne contamination (by maintaining slightly negative pressure) during Loss of Coolant Accidents (LOCA). The two subsystems are:

1. A once through filtration exhaust system, designed to contain particulate and gaseous contamination and prevent it from being released from the building in accordance with 10CFR20, and

2. A once through air supply system, designed to deliver outside air into the building to maintain building temperatures within acceptable limits. For the purposes of satisfying the Technical Specification LCO, one supply fan must be administratively removed from service such that the fan will not auto-start on an actuation signal; however, the supply fan must be OPERABLE with the exception of this administrative control.

These systems operate during normal and emergency plant modes. Additionally, the system provides a flow path for containment purge supply and exhaust during Modes 5 and 6. Either the Containment Purge system or the Auxiliary Building Ventilation System with suction from the containment atmosphere, with associated radiation monitoring will be available whenever movement of irradiated fuel is in progress in the containment building and the equipment hatch is open. If for any reason, this ventilation requirement can not be met, movement of fuel assemblies within the containment building shall be discontinued until the flow path(s) can be reestablished or close the equipment hatch and personnel airlocks.

Appropriate filtration surveillances are contained in the Updated Final Safety Analysis Report (UFSAR) Section 9.4.2.4, Test and Inspections. Auxiliary Building exhaust air filtration system functionality is not required to meet LCO 3.7.7.

The ventilation exhaust consists of three 50% capacity fans that are powered from vital buses. The fans are designed for continuous operation, to control the Auxiliary Building pressure at –0.10” Water Gauge with respect to atmosphere.

The ventilation supply consists of two 100% capacity fans that are powered from vital buses, and distribute outdoor air to the general areas and corridors of the building through associated ductwork.

3/4.7.7 AUXILIARY BUILDING EXHAUST AIR FILTRATION SYSTEM (cont'd)

AUXILIARY BUILDING VENTILATION ALIGNMENT MATRIX
NORMAL VENTILATION (Normal plant operations)*

Any two of the three exhaust fans and either of the two supply fans:

- The normal alignment is two exhaust fans and one supply fan. During cooler seasons, and with the absence of the system heating coils, it may be required to limit the amount of colder outside air entering the building. In this case, it is acceptable to secure both supply fans from operation and reduce the number of operating exhaust fans to one. There is sufficient capacity with the single exhaust fan to maintain the negative pressure within the auxiliary building boundary.

EMERGENCY VENTILATION (Emergency plant operation)

At least two of the three exhaust fans and either one of the two supply fans.

Note: During a Safety Injection (SI) all three exhaust fans and one of the supply fans will start. This is acceptable and will maintain the boundary pressure while supplying the required cooling to the building. Should access/egress become difficult with the three exhaust fans running, one of the exhaust fans should be secured.

OPERABILITY of the Auxiliary Building Ventilation System ensures that air, which may contain radioactive materials leaked from ECCS equipment following a LOCA, is monitored prior to release from the plant via the plant vent.
Operation of this system and the resultant effect on offsite and control room dose calculations was assumed in the accident analyses. ABVS is discussed in UFSAR Section 9.4.2.

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.9 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from the program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they were installed, would have no adverse effect on any safety-related system.

The program for examination, testing and service life monitoring for snubbers is required to be performed in accordance with ASME BPV Code, Section XI or the OM Code and the applicable addenda as required by 10 CFR 50.55a(g) or 10 CFR 50.55a(b)(3)(v), except where the NRC has granted specific written relief, pursuant to 10 CFR 50.55a(g)(6)(i), or authorized alternatives pursuant to 10 CFR 50.55a(a)(3).

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3/4.7.10 CHILLED WATER SYSTEM - AUXILIARY BUILDING SUBSYSTEM

The OPERABILITY of the chilled water system ensures that the chilled water system will continue to provide the required normal and accident heat removal capability for the control room area, relay rooms, equipment rooms, and other safety related areas. Verification of the actuation of each automatic valve on a Safeguards Initiation signal includes actuation following receipt of a Safety Injection signal.

The Auxiliary Building Chilled Water (AB CH) systems can be operated in three possible LCO configurations:

1. Three Chillers Required (LCO 3.7.10.a)
2. Two Chillers Required (LCO 3.7.10.b)
3. Units Cross-Tied (LCO 3.7.10.c)

Three Chillers Required Configuration:

Removal of non-essential heat loads from the chilled water system in the event one chiller is inoperable ensures the remaining heat loads are within the heat removal capacity of the two operable chillers.

Removal of non-essential heat loads from the chilled water system in the event two chillers are inoperable and aligning the CREACs to the maintenance mode ensures the remaining heat loads are within the heat removal capacity of the operable chiller.

During chiller testing, operator actions can take the place of automatic actions

During Modes 5 and 6 and during movement of irradiated fuel assemblies, chilled water components do not have to be considered inoperable solely on the basis that the backup emergency power source, diesel generator, is inoperable. This is consistent with Technical Specification 3.8.1.2 which only requires two operable diesel generators.

Two Chillers Required Configuration:

In Two Chiller configuration the analyses demonstrate the system will continue to provide required cooling capability to the control room and safety related areas during normal operation and in the event of an accident in conjunction with a single failure. The analyses for Two Chiller configuration were performed with both trains of Control Room Emergency Air Conditioning (CREACS) operable and one chiller operating in each unit. This configuration accounts for one of the two required chillers in a unit being out of service and an accident and single failure (loss of chiller) in the opposite unit. The restrictions for entering Two Chiller configuration ensure that the heat loads are within the heat removal capacity of the remaining operable chiller. The heat removal capacity of the chiller is based on the service water and outside air temperatures present during the period of November 1st through April 30th. Removal of the Emergency Control Air Compressor (ECAC) from the CH system ensures that the heat load is within the capacity of the remaining chiller.

If one unit is in the Two-Chiller configuration (LCO 3.7.10.b) and the other unit is in the Three Chiller configuration (LCO 3.7.10.a), CREACS single filtration train alignment is allowed with the unit that is in Three Chiller configuration supplying the CREACS train. Additionally, nonessential heat loads must be isolated from the chilled water system on BOTH Units. Alignment of the single CREACS train to the unit in the Two-Chiller configuration is not permitted.

When entering LCO 3.7.10.b, the third chiller must have CH flow isolated to prevent recirculation of cooling water flow through the non-operating chiller. When restoring from LCO 3.7.10.b for transitioning to the Three Chiller configuration, the third chiller may be un-isolated under administrative controls. The administrative controls will require that an operator be dedicated during restoration activities to re-isolate the chiller, if necessary, in the event an accident occurs during the restoration activities.

The loss of the 2 required chillers requires the unit that has the lost the chillers to commence a controlled shutdown (or suspend CORE ALTERATIONS and movement of irradiated fuel assemblies if in MODES 5 or 6 or during the movement of irradiated fuel) and transition the CREACS to single filtration operation with the opposite unit supplying the CREACS train unless both units transition to the Cross-Tied configuration. In the event that the Cross-tied configuration cannot be implemented or the transition to CREACS single filtration train alignment cannot be implemented, both units will commence a controlled shutdown (or suspend CORE ALTERATIONS and movement of irradiated fuel assemblies if in MODES 5 or 6 or during the movement of irradiated fuel).

Required operating conditions will be verified every 24-hours (SR 4.7.10.d) when in the Two-Chiller configuration.

Cross-Tied Configuration:

In Cross-tie configuration the analyses demonstrate the system will continue to provide required cooling capability to the control room and safety related areas during normal operation and in the event of an accident in either unit. The supporting calculations were performed assuming that one of the required chillers is unavailable due to either a single failure or being out of service (two chillers remaining).

The analyses for Cross-Tied configuration determined that both train of CREACS must be operable. With only a single train of CREACS operable, the remaining CREACS cooling coil cannot maintain the control room envelope temperatures within acceptable limits. Therefore, entry into CH Cross-Tied configuration is only allowed when both trains of CREACS are operable. A note is added to TS 3.7.6 Action a to alert operators that CREACS single filtration operation is not permitted if the units are in the CH Cross-tied configuration.

The restrictions for entering the Cross-Tied configuration ensure that the heat loads are within the heat removal capacity of the remaining two operable chillers. The heat removal capacity of the chillers is based on the service water and outside air temperatures present during the

period of November 1st through April 30th. Removal of both units' ECACs and both units' non-essential heat loads from the CH system ensures that the heat load is within the capacity of the remaining chillers.

When restoring from LCO 3.7.10.c, the cross-tie valve can be closed under administrative controls. The administrative controls will require that an operator be dedicated during restoration activities to re-open the cross-tie valve, if necessary, in the event an accident occurs during the restoration activities.

If two Chillers become inoperable in Cross-Tie configuration then both units must commence a controlled shutdown (or suspend CORE ALTERATIONS and movement of irradiated fuel assemblies if in MODES 5 or 6 or during the movement of irradiated fuel).

Required operating conditions will be verified every 24-hours (SR 4.7.10.e) when in the Cross-Tied configuration.

3/4.7.11 FUEL STORAGE POOL BORON CONCENTRATION

In the Maximum Density Rack (MDR) design, the spent fuel storage pool is divided into two separate and distinct regions. Region 1, with 300 storage positions, is designed to accommodate new fuel with a maximum enrichment of 4.25 wt% U-235. Unirradiated and irradiated fuel with initial enrichments up to 5.0 wt% U-235 can also be stored in Region 1 with some restrictions. These restrictions are stated in TS 3/4.7.12. Region 2, with 1332 storage positions, is designed to accommodate unirradiated and irradiated fuel with stricter controls as compared to Region 1. These controls are also stated in TS 3/4.7.12.

The water in the spent fuel storage pool 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. The double contingency principle discussed in ANSI N-16.1-1975 and the USNRC letter of April 14, 1978, to all Power Reactor Licensees – OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications (Accession # 7910310568) allows credit for soluble boron under other abnormal or accident conditions, consistent with postulated accident scenarios. For example, the most severe accident scenario is associated with the abnormal location of a fresh fuel assembly of 5.0 wt% enrichment which could, in the absence of soluble poison, result in exceeding the design reactivity limitation (k_{eff} of 0.95). This could occur if a fresh fuel assembly of 5.0 wt% enrichment were to be inadvertently loaded into a Region 1 or Region 2 storage cell otherwise filled to capacity. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Calculations for the worst case configuration confirmed that 800 ppm soluble boron (includes an appropriate allowance for boron concentration measurement uncertainty)is adequate to compensate for a mis-located fuel assembly. Subcriticality of the MDR with no movement of assemblies is achieved without credit for soluble boron and by controlling the location of each assembly in accordance with TS 3/4.7.12. Prior to movement of an assembly, it is necessary to verify the fuel storage pool boron concentration is within limit in accordance with TS 3/4.7.11.

Most postulated abnormal conditions or accidents in the spent fuel pool do not result in an increase in the reactivity of either MDR region. For example, an event that results in an increase in spent fuel pool temperature or a decrease in water density will not result in a reactivity increase. An event that results in the spent fuel pool cooling down below normal conditions does not impact the criticality analysis since the analysis assumes a water temperature of 4°C. This assures that the reactivity will always be lower over the expected range of water temperatures.

3/4.7.11 FUEL STORAGE POOL BORON CONCENTRATION (continued)

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 exceeding limits in both regions. The postulated accidents are basically of three types. The first type of postulated accident is an abnormal location of a fuel assembly, the second type of postulated accident is associated with lateral rack movement, and the third type of postulated accident is a dropped fuel assembly on the top of the rack. The dropped fuel assembly and the lateral rack movement have been previously shown to have negligible reactivity effects (<0.0001 δk). The misplacement of a fuel assembly could result in Keff exceeding the 0.95 limit. However, the negative reactivity effect of a minimum soluble boron concentration of 600 ppm compensates for the increased reactivity caused by any of the postulated accident scenarios. The accident analyses are summarized in the FSAR Section 9.1.2.

The determination of 600 ppm has included the necessary tolerances and uncertainties associated with fuel storage rack criticality analyses. To ensure that soluble boron concentration measurement uncertainty is appropriately considered, additional margin is incorporated into the limiting condition for operation. As such, increasing the minimum required boron concentration in the fuel storage pool to 800 ppm conservatively covers the expected range of boron reactivity worth along with allowances associated with boron measurements.

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). The fuel storage pool boron concentration is required to be greater than or equal to 800 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool, until a complete spent fuel storage pool verification has been performed following the last movement of fuel assemblies in the spent fuel storage pool. 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.

3/4.7.11 FUEL STORAGE POOL BORON CONCENTRATION (continued)

The Required Actions are modified indicating that LCO 3.0.3 does not apply. Storage of fuel assemblies and the boron concentration in the spent fuel storage pool are independent of reactor operation. Therefore TS 3/4 3.7.11 and TS3/ 4 3.7.12 include the exception to LCO 3.0.3 to preclude an inappropriate reactor shutdown. When the concentration of boron in the fuel storage pool 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 movement of fuel assemblies. Alternatively, beginning a verification of the fuel storage pool 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 fuel assemblies in the spent fuel pool while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving fuel assemblies in the spent fuel pool 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.

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

3/4.7.12 FUEL ASSEMBLY STORAGE IN THE SPENT FUEL POOL

In the Maximum Density Rack (MDR) design, the spent fuel storage pool is divided into two separate and distinct regions. Region 1, with 300 storage positions, is designed to accommodate new fuel with a maximum enrichment of 4.25 wt% U-235. Unirradiated and irradiated fuel with initial enrichments up to 5.0 wt% U-235 can also be stored in Region 1 with some restrictions. These restrictions are stated in TS 3/4.7.12. Region 2, with 1332 storage positions, is designed to accommodate unirradiated and irradiated fuel with stricter controls as compared to Region 1. These controls are also stated in TS 3/4.7.12.

The water in the spent fuel storage pool 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. The double contingency principle discussed in ANSI N-16.1-1975 and the USNRC letter of April 14, 1978, to all Power Reactor Licensees – OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications (Accession # 7910310568) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the abnormal location of a fresh fuel assembly of 5.0 wt% enrichment which could, in the absence of soluble poison, result in exceeding the design reactivity limitation (k_{eff} of 0.95). This could occur if a fresh fuel assembly of 5.0 wt% enrichment were to be inadvertently loaded into a Region 1 or Region 2 storage cell otherwise filled to capacity, for any of the configurations. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Calculations for the worst case configuration confirmed that 800 ppm soluble boron (includes an appropriate allowance for boron concentration measurement uncertainty) is adequate to compensate for a mis-located fuel assembly. Safe operation of the MDR with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with TS 3/4.7.12. Prior to movement of an assembly into a fuel assembly storage location in Region 1 or Region 2, it is necessary to perform SR 4.7.11 and either SR 4.7.12.1 or SR 4.7.12.2. In summary, before moving an assembly into the storage racks it is necessary to:

- validate that its final location meets the criticality requirements;
- and since there is a potential to misload the assembly, we need to ensure that the Fuel Storage Pool boron concentration is greater than the minimum required to preclude exceeding criticality limits prior to moving.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

3/4.7.12 FUEL ASSEMBLY STORAGE IN THE SPENT FUEL POOL (CONTINUED)

The restrictions on the placement of fuel assemblies within the spent fuel pool in accordance with TS 3/4.7.12, in the accompanying LCO, ensures the k_{eff} of the spent fuel storage pool will always remain < 0.95, assuming the pool to be flooded with unborated water.

This LCO applies whenever any fuel assembly is stored in Region 1 or Region 2 of the fuel storage pool.

The Required Actions are modified indicating that LCO3.0.3 does not apply. Storage of fuel assemblies and the boron concentration in the spent fuel storage pool are independent of reactor operation. Therefore TS 3/4.3.7.11 and TS 3/4.3.7.12 include the exception to LCO 3.0.3 to preclude an inappropriate reactor shutdown. When the configuration of fuel assemblies stored in Region 1 or Region 2 of the spent fuel storage pool is not in accordance with TS 3/4.7.12, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with TS 3/4.7.12. If unable to move fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move 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.

The SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with TS 3/4.7.12 in the accompanying LCO.

3/4.7.13 Main Feedwater Isolation Valves (FIVs), Main Feedwater Regulating Valves (FRVs), FRV Bypass Valves (FRVBVs), and Steam Generator Feedwater Pump (SGFP) Turbine Steam Stop Valves

The OPERABILITY of the FIVs (BF13s), FRVs (BF19s), FRVBVs (BF40s) and SGFP turbine steam stop valves (MS43s and RS15s) ensures that the valves will be capable of performing their intended safety function. The safety function of these valves is to rapidly close following: (1) a steam line or feedwater line rupture, thereby limiting the Reactor Coolant System cooldown and limiting the total energy release to the containment; or (2) a feedwater system malfunction, thereby limiting Reactor Coolant System cooldown.

The analysis of excessive RCS heat removal due to a feedwater system malfunction (UFSAR Section 15.2.1 0) assumes full opening of one or more FRVs, due to a control system malfunction or operator error, resulting in a step increase in feedwater flow to one or more steam generators. The analysis assumes a feedwater isolation signal is generated by a high-high steam generator level or a safety injection (SI) signal. Feedwater isolation is assumed to occur as a result of the FIV(s) closing in 32 seconds and failure of the FRV(s) and associated FRVBV(s) to close as a result of the feedwater isolation signal. The trip of the SGFPs is not credited in the feedwater system malfunction analysis.

Rupture of a steam line (UFSAR Section 15.4.2) is analyzed to determine the response of the reactor core and to determine the resulting mass and energy releases. Two separate analyses are performed since conservative assumptions for the core response analysis are different than the conservative assumptions for the mass and energy release analysis.

The core response analysis credits feedwater isolation as a result of the safety injection signal which results in a feedwater isolation signal. For the steam generators with the non-faulted steam line, feedwater isolation is assumed to occur as a result of closure of all FRVs and FRVBVs in 10 seconds following receipt of the SI signal. For the steam generator with the faulted steam line, the FRV is assumed to fail with feedwater isolation achieved by closure of the FIV in 32 seconds following receipt of the SI signal.

The mass and energy release analysis assumes feedwater isolation occurs as a result of the SI signal which generates the feedwater isolation signal. The most limiting case for mass addition to the containment assumes failure of the FRV to close in the loop with the faulted steam line. Feedwater isolation occurs as a result of closure of the FIVs in 32 seconds and tripping of the SGFPs in 7 seconds. Reduction of feedwater flow due to SGFP coast down and closure of the FIV is credited in the containment analysis.

Rupture of a feedwater line between the feedwater stop-check valve and the steam generator (UFSAR Section 15.4.3) is analyzed to determine the response of the reactor core. The feedwater line break could cause either a reactor coolant system (RCS) cooldown or a heat up depending on the size of the rupture.

The RCS cooldown for a feedwater line rupture is bounded by the analysis for a steam line rupture. For the RCS heatup analysis, main feedwater to all steam generators is assumed to stop at the time of the feedwater line rupture due to the feedwater spilling out the break. A feedwater isolation signal is generated as a result of the safety injection signal and is accomplished by closure of the FRVs and FRVBVs in 10 seconds following receipt of the SI signal.

The mass and energy release that would result from a rupture of a main feedwater line inside containment is bounded by the analysis of the rupture of a main steam line.

The APPLICABILITY of this specification is MODES 1, 2, and 3, except when:

- a FIV or FRV and FRVBV valve are closed and deactivated or the main feedwater line is isolated by a closed manual valve; or
- the SGFP turbine steam stop valve is closed and deactivated or the steam supply to the SGFP turbine is isolated, or the SGFP discharge to the steam generators is isolated.

The basis for the mode applicability is that in MODES 1 and 2 there is significant mass and energy in the RCS and steam generators and in MODE 3 there may be significant mass and energy in the RCS and steam generators. With significant mass and energy in the RCS and steam generators, the valves are needed for isolation of the steam generators in the event of a secondary system pipe rupture. The mode applicability is modified by exception based upon the impacted valve or SGFP being placed in its required accident-analysis assumed position or the flow path being isolated such that the credited accident analysis function has already been completed.

The ACTION statements for an inoperable FIV, FRV, or FRVBV require that action must be taken to restore the affected valves to OPERABLE status, or to close or isolate the inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function. The 72 hour action 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 feedwater flow paths. The 72 hour action time is reasonable, based on operating experience.

The ACTION statement for an inoperable SGFP turbine steam stop valve requires that action must be taken to restore the affected valves to OPERABLE status, or isolate the associated steam supply to the SGFP, or isolate the associated SGFP feedwater flow path within 72 hours. When the SGFP steam stop valves are closed or isolated, or the SGFP feedwater flow path is isolated, the required safety function has been completed. The 72 hour action 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 action time is reasonable, based on operating experience.

Inoperable 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 action 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.

Separate ACTION entry is allowed for each inoperable valve unless there is a loss of feedwater isolation capability for a flow path. Redundant components in the flow path would perform the feedwater isolation function.

With either (1) a FRV or FRVBV and FIV inoperable, or (2) SGFP turbine steam stop valve (resulting in a loss of SGFP trip function) and FRV or FRVBV inoperable, 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 FIV, FRV, FRVBV, or SGFP turbine steam stop valve, or otherwise isolate the affected flow path. With both a SGFP turbine steam stop valve and FIV inoperable, the FRV and FRVBV will operate automatically to provide feedwater isolation for the flow path.

If the FIV(s), FRV(s), FRVBV(s) and SGFP turbine steam stop valves cannot be restored to OPERABLE status, or closed, or the flow path isolated within the associated allowed outage 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 HOT STANDBY (MODE 3) within 6 hours and in HOT SHUTDOWN (MODE 4) within the following 6 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.

SR 4.7.13.1 verifies that the closure time of each FIV, FRV, FRV bypass and SGFP turbine steam stop valve is within the limit in the Technical Requirements Manual and is within that assumed in the accident and containment analyses. This SR also verifies the valve closure time is in accordance with the Inservice Testing Program. The Frequency for this SR is in accordance with the Inservice Testing Program.

SR 4.7.13.2 verifies that each FIV, FRV, FRV bypass and SGFP turbine steam stop valve can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

3/4.8.1 and 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility, and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR Part 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the accident analyses and are based upon maintaining at least two independent sets of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of one onsite A.C. source.

When a system or component is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may still be considered OPERABLE, provided the appropriate Actions of 3.8.1.1.a.2, b.2 or d.2 are satisfied.

Action 3.8.1.1.a.2, which only applies if the train cannot be powered from an offsite source, 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 safety function of critical redundant required systems. Failure of a single offsite circuit will generally not, by itself, cause any equipment to lose normal AC power. Action 3.8.1.1.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 safety function of critical systems. Action 3.8.1.1.d.2, 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.

These systems are powered from the independent AC electrical power train. However, redundant required systems or components credited by this specification are not necessarily powered from AC electrical sources. For example, the single train turbine-driven auxiliary feedwater pump is redundant to the two motor-driven pumps. Redundant required system or component failures consist of inoperable equipment associated with a train, redundant to the train that has an inoperable DG or offsite power.

LCO 3.0.4.b is not applicable to an inoperable DG. There is an increased risk associated with entering a MODE or other specified condition in the Applicability with an inoperable DG. The provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

The completion time for these actions 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, starting only on discovery that both:

a. One train has no offsite power supplying its loads, one DG is inoperable or two required offsite circuits are inoperable; and

b. A required system or component on the other train is inoperable.

If at any time during these conditions a redundant required system or component subsequently becomes inoperable, this completion time begins to be tracked. Discovering no offsite power to one train of the onsite Class 1E Electrical Power Distribution System, or one required DG inoperable, coincident with one or more inoperable required support or supported systems

or components that are associated with the other train that has power, results in starting the completion times for the Action. The specified time is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE AC supplies (one offsite circuit and three DGs for Condition (a), two offsite circuits and two DGs for Condition (b), or three DGs for Condition (d)) are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required system or component's function may have been lost; however, function has not been lost. The completion time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required system or component. Additionally, the 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. The completion time for Condition d (loss of both offsite circuits) is reduced to 12 hours from that allowed for one train without offsite power (Action 3.8.1.1.a.2). The rationale is that Regulatory Guide 1.93 allows a completion time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required system or component failure exists, this assumption is not the case, and a shorter completion time of 12 hours is appropriate.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Applicability of specifications 3.8.2.2, 3.8.2.4, and 3.8.2.6 includes the movement of irradiated fuel assemblies. This will insure adequate electrical power is available for proper operation of the fuel handling building ventilation system during movement of irradiated fuel in the spent fuel pool.

An offsite circuit would be considered inoperable if it were not available to one required train. Although two trains are required by LCOs 3.8.2.2 and 3.8.2.4, the one train with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and irradiated 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.

With the offsite circuit or diesel generator not available to all required trains, the option exists to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With both required diesel generators inoperable, the minimum required diversity of AC power sources is not available. Therefore, it is required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required shutdown margin or boron concentration. Suspending positive reactivity additions that could result in failure to meet the minimum shutdown margin or boron concentration limit is required to assure continued safe operation.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are based upon the recommendations of Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. Regulatory Guide 1.108 criteria for determining and reporting valid tests and failures, and accelerated diesel generator testing, have been superseded by implementation of the Maintenance Rule for the diesel generators per 10CFR50.65. In addition to the Surveillance Requirements of 4.8.1.1.2, diesel preventative maintenance is performed in accordance with procedures based on manufacturer's recommendations with consideration given to operating experience.

The minimum voltage and frequency stated in the Surveillance Requirements (SR) are those necessary to ensure the Emergency Diesel Generator (EDG) can accept Design Basis Accident (DBA) loading while maintaining acceptable voltage and frequency levels. Stable operation at the nominal voltage and frequency values is also essential in establishing EDG OPERATILITY, but a time constraint is not imposed. The lack of a time constraint is based on the fact that a typical EDG will experience a period of voltage and frequency oscillations prior to reaching steady state operation if these oscillations are not dampened out by load application. In lieu of a time constraint in the SR, controls will be provided to monitor and trend the actual time to reach stable operation within the band as a means of ensuring there is no voltage regulator or governor degradation that could cause an EDG to become inoperable.

"Standby condition" for the purpose of defining the condition of the engine immediately prior to starting for surveillance requirements requires that the lube oil temperature be between 100 °F and 170 °F. The minimum lube oil temperature for an OPERABLE diesel is 100 °F.

The thirteen second time requirement for the Emergency Diesel Generator to reach rated voltage and frequency was originally based on a Westinghouse assumption of fifteen seconds that included the delay time between the occurrence of the incident and the application of electrical power to the first sequenced safeguards pump (BURL-3011, dated November 13, 1974) and included an instrument response time of two seconds (BURL-1531, dated July 27, 1970). The times specified in UFSAR Section 15.4 bound the thirteen seconds specified in the TS.

The narrower band for frequency specified for testing performed in steady state isochronous operation will ensure the EDG will not be run in an overloaded condition (steady state) during accident conditions. Steady state is assumed to be achieved after one minute of operation in the isochronous mode with all required loads sequenced on the bus.

The narrower band for steady state voltage is specified for operation when the EDG is not synchronized to the grid to ensure the voltage regulator will protect driven equipment from over-voltages during accident conditions. Procedural controls will ensure that equipment voltages are maintained within acceptable limits during testing when paralleled to the grid.

The wider band for frequency is appropriate for testing done with the governor in the droop mode. Likewise the wider band for voltage is appropriate when paralleled to the grid.

All voltages and frequencies specified in SR 4.8.1.1.2 are representative of the analytical values and do not account for postulated instrument inaccuracy. Instrument inaccuracies for EDG voltage and frequency are administratively controlled.

Preventive maintenance includes those activities (including pro-test inspections, measurements, adjustments and preparations) performed to maintain an otherwise OPERABLE EDG in an OPERABLE status. Corrective maintenance includes those activities required to correct a condition that would cause the EDG to be inoperable.

Surveillance requirement 4.8.1.2 is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of the surveillance requirement, and to preclude de-energizing a required ESF bus or disconnecting a required offsite circuit during performance of surveillance requirements. With limited AC sources available, a single event could compromise both the required circuit and the DG. It is the intent that these surveillance requirements must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit are required to be OPERABLE. During Startup, prior to entering Mode 4, the surveillance requirements are required to be completed if the surveillance frequency has been exceeded or will be exceeded prior to the next scheduled shutdown.

3/4.8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.. Each manufacturer's molded case circuit breakers and lower voltage circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of molded case or lower voltage circuit breakers, it is necessary to further divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

Containment penetration conductor overcurrent protective device information is provided in the UFSAR.

3/4.9.1 BORON CONCENTRATION

The limit on the boron concentration of the Reactor Coolant System (RCS), the refueling cavity, the fuel storage pool and 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 Core Operating Limits Report (COLR). Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of Keff $\leq$ 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.

General Design Criterion 26 of 10CFR 50, Appendix A requires that two independent reactivity control systems of different design principles be provided. 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 through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps. The fuel storage pool is also adjusted to the refueling boron concentration specified in the COLR.

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 TS 3/4.9.8, "Residual Heat Removal (RHR) and Coolant Circulation - All Water levels, " and "Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

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 Keff of the core will remain $\leq$ 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 pool, 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 RCS boron concentration satisfies Criterion 2 10CFR50.36(c)(2)(ii).

The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal, the fuel storage pool and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures that a core Keff $\leq$ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

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 Keff $\leq$ 0.95. A note to this LCO modifies the Applicability. The note states that the limits on boron concentration are only applicable to the refueling canal, the fuel storage pool and the refueling cavity when those volumes are connected to the Reactor Coolant System. When the refueling canal, the fuel storage pool and the refueling cavity are isolated from the RCS, no potential path for boron dilution exists. Above MODE 6, LCOs 3.1.1.1 and 3.1.1.2 ensure that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

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, the refueling canal, the fuel storage pool 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. Operations that individually add limited positive reactivity (e.g. temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action.

In addition to immediately suspending CORE ALTERATIONS and 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. 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.

The Surveillance Requirement (SR) ensures that the coolant boron concentration in the RCS, and connected portions of the refueling canal, the fuel storage pool and the refueling cavity, is within the COLR limits. The boron concentration of the coolant in each required volume is determined periodically by chemical analysis. Prior to reconnecting portions of the refueling canal, the fuel storage pool or the refueling cavity to the RCS, this SR must be met per SR 4.0.4. If any dilution activity has occurred while the cavity or canal was disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

3/4.9.2.1 UNBORATED WATER SOURCE ISOLATION VALVES

During MODE 6 operations, all isolation valves for the reactor makeup water sources containing unborated water that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position.

Securing the required valves in the closed position during refueling operations ensures that the valves cannot be inadvertently opened, and prevents the flow of unborated water to the filled portion of the RCS. This action precludes the possibility of an inadvertent boron dilution event occurring during MODE 6 refueling operations. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution event in accordance with the Standard Review Plan (NUREG-0800, Section 15.4.6) is not required for MODE 6.

If any required valve is found not secured in the closed position, there is a potential of having a diluted boron concentration in the RCS. Immediately suspend CORE ALTERATIONS, and initiate actions to secure the valve in the closed position. Surveillance Requirement 4.9.1 must be performed to demonstrate that the required boron concentration exists. The 4 hour completion time is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

Surveillance Requirement 4.9.2.1 demonstrates through a system walkdown that the required valves are closed. The surveillance frequency is controlled under the Surveillance Frequency Control Program.

3/4.9.2.2 INSTRUMENTATION

The source range neutron flux monitors are used during refueling operations to determine the core reactivity condition. Two OPERABLE source range neutron flux monitors are required to

alert the operator to unexpected changes in core reactivity, such as a boron dilution event. This ensures that redundant monitoring capability is available to detect changes in core reactivity. Based on isolating all boron dilution paths per LCO 3.9.2.1, only the source range neutron flux monitor visual indication in the control room is required for OPERABILITY.

Any combination of NIS source range neutron flux monitors and/or Gamma-Metrics post-accident neutron flux monitors may be used to satisfy the LCO. Two of the four total source range neutron flux monitors are required to be OPERABLE.

With only one required source range neutron flux monitor OPERABLE, redundancy has been lost. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation.

With no required source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. With no source range neutron flux monitor OPERABLE, there is no direct means of detecting changes in core reactivity. However, since 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 confirmed by performing Surveillance Requirement 4.9.1 to ensure that the required boron concentration exists and adequate shutdown margin is maintained.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. The 80-hour decay time (LAR S08-01) is consistent with the assumptions used in the fuel handling accident analyses and the resulting dose calculations using the Alternative Source Term described in Reg. Guide 1.183.

The minimum requirement for reactor subcriticality also ensures that the decay time is consistent with that assumed in the Spent Fuel Pool cooling analysis. The calendar based restrictions are established for the actual movement of irradiated fuel; i.e., movement cannot commence in the October 15th through May 15th window unless at least 80 hours has elapsed since subcriticality was achieved. The 80 hour clock can start prior to October 15 but must end in the October 15th – May 15th window for the 80 hour criteria to be applicable. Similarly, fuel movement between May 16th and October 14th cannot commence unless at least 168 hours has elapsed since subcriticality was achieved.

Delaware River water average temperature between October 15th and May 15th is determined from historical data taken over 30 years. The use of 30 years of data to select maximum temperature is consistent with Reg. Guide 1.27, “Ultimate Heat Sink for Nuclear Power Plants”.

A core offload has the potential to occur during both applicability time frames. In order not to exceed the analyzed Spent Fuel Pool cooling capability to maintain the water temperature below 180°F, two decay time limits are provided. In addition, PSEG has developed and implemented a Spent Fuel Pool Integrated Decay Heat Management Program as part of the Salem Outage Risk Assessment. This program requires a pre-outage assessment of the Spent Fuel Pool heat loads and heat-up rates to assure available Spent Fuel Pool cooling capability prior to offloading fuel.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

During movement of irradiated fuel assemblies within containment the requirements for containment building penetration closure capability and OPERABILITY ensure that a release of fission product radioactivity within containment will not exceed the guidelines and dose calculations described in Reg Guide 1.183, Alternative Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Plants. In MODE 6, the potential for containment pressurization as a result of an accident is not likely. Therefore, the requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements during movement of irradiated fuel assemblies within containment are referred to as “containment closure” rather than containment OPERABILITY. For the containment to be OPERABLE, CONTAINMENT INTEGRITY must be maintained. Containment closure means that all potential release paths are closed or capable of being closed. Closure restrictions include the administrative controls to allow the opening of both airlock doors and the equipment hatch during fuel movement provided that: 1) the equipment inside door or an equivalent closure device installed is capable of being closed with four bolts within 1 hour by a designated personnel; 2) the airlock doors are capable of being closed within 1 hour by designated personnel, 3) either the Containment Purge System or the Auxiliary Building Ventilation System taking suction from the containment atmosphere are operating and 4) the plant is in Mode 6 with at least 23 feet of water above the reactor pressure vessel flange.

Administrative requirements are established for the responsibilities and appropriate actions of the designated personnel in the event of a Fuel Handling Accident inside containment. These requirements include the responsibility to be able to communicate with the control room, to ensure that the equipment hatch is capable of being closed, and to close the equipment hatch and personnel airlocks within 1 hour in the event of a fuel handling accident inside containment. These administrative controls ensure containment closure will be established in accordance with and not to exceed the dose calculations performed using guidelines of Regulatory Guide 1.183.

The containment serves to limit the 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 10CFR100 and Reg Guide 1.183, Alternative Source Term, as applicable. 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 or out of containment. During movement of irradiated fuel assemblies within containment can be open provided that: 1) it is capable of being closed with four bolts within 1 hour by designated personnel, 2) either the Containment Purge System or the Auxiliary Building Ventilation System taking suction from the containment atmosphere are operating and 3) the plant is in Mode 6 with at least 23 feet of water above the reactor pressure vessel flange. Good engineering practice dictates that the bolts required by the LCO are approximately equally spaced.

An equivalent closure device may be installed as an alternative to installing the Containment Equipment Hatch inside door with a minimum of four bolts. Such a closure device may provide penetrations for temporary services used to support maintenance activities inside containment at times when containment closure is required; and may be installed in place of the Containment Equipment Hatch inside door or outside door. Penetrations incorporated into the design of an equivalent closure device will be considered a part of the containment boundary and as such will be subject to the requirements of Technical Specification 3/4.9.4. Any equivalent closure device used to satisfy the requirements of Technical Specification 3/4.9.4.a will be designed, fabricated, installed, tested, and utilized in accordance with established procedures to ensure that the design requirements for the mitigation of a fuel handling accident during refueling operations are met. In case that this equivalent closure device is installed in lieu of the equipment hatch inside door, the same restrictions and administrative controls apply to ensure closure will take place within 1 hour following a Fuel Handling Accident inside containment.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during operation in MODES 1, 2, 3, and 4 as specified in LCO 3.6.1.3, "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 and frequent containment entry is necessary, the air lock interlock mechanism may be disabled. This allows both doors of an airlock to remain open for extended periods. During movement of irradiated fuel assemblies within containment, containment closure may be required; therefore, the door interlock mechanism may remain disabled, and both doors of each containment airlock may be open if: 1) At least one door of each airlock is capable of being closed within 1 hour by dedicated personnel,) 2) either the Containment Purge System or the Auxiliary Building Ventilation System taking suction from the containment atmosphere are operating and 3) The plant is in Mode 6 with at least 23 feet of water above the reactor pressure vessel flange.

In the postulated FHA, the revised dose calculations performed using RG 1.183 criteria, do not assume automatic containment purge isolation thus allowing for continous monitoring of containment acitivity until the release pathways are isolated. If required, manual isolation of containment purge can be initiated from the control room.

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 may include the use of a material that can provide a temporary atmospheric pressure, ventilation barrier. Any equivalent method used to satisfy the requirements of Technical Specification 3/4.9.4.c.1 will be designed, fabricated, installed, tested, and utilized in accordance with established procedures to ensure that the design requirements for the mitigation of a fuel handling accident during refueling operations are met.

The surveillance requirement 4.9.4.2 demonstrates that the necessary hardware, tools, and equipment are available to close the equipment hatch. The surveillance is performed once per refueling prior to the start of movement of irradiated fuel assemblies within the containment. This surveillance is only required to be met when the equipment hatch is open.

3/4.9.5 COMMUNICATIONS

Deleted.

3/4.9.6 MANIPULATOR CRANE

Deleted.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

Deleted.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirements that at least one residual heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification. A minimum flow rate of 1000 gpm is required. Additional flow limitations are specified in plant procedures, with the design basis documented in the Salem UFSAR. These flow limitations address the concerns related to vortexing and air entrapment in the Residual Heat Removal system, and provide operational flexibility by adjusting the flow limitations based on time after shutdown. The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability.

For support systems: Service Water (SW) and Component Cooling (CC), component redundancy is necessary to ensure no single active component failure will cause the loss of Decay Heat Removal. One piping path of SW and CC is adequate when it supports both RHR loops. The support systems needed before entering into the desired configuration (e.g., one service water loop out for maintenance in Modes 5 and 6) are controlled by procedures, and include the following:

- A requirement that the two RHR, two CC and two SW pumps, powered from two different vital buses be kept operable
- A listing of the active (air/motor operated) valves in the affected flow path to be locked open or disable.

Note that four filled reactor coolant loops, with at least two steam generators with at least their secondary side water level greater than or equal to 5% (narrow range), may be substituted for one residual heat removal loop. This ensures that single failure does not cause a loss of decay heat removal.

With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 (Not Used)

3/4.9.10 and 3/4/9/11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.12 FUEL HANDLING AREA VENTILATION SYSTEM

The operability of the Fuel Handling Area Ventilation System during movement of irradiated fuel ensures that a release of fission product radioactivity within the Fuel Handling Building will not exceed the guidelines and dose calculations described in Reg. Guide 1.183, Alternative Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Reactors.

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure control rod worth, and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the Reactor Coolant System T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the reactor core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is, at times, necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not be allowed by Specification 3.1.3.5 which may, in turn, cause the RCS T_{avg} to fall slightly below the minimum temperature of Specification 3.1.1.4.

3/4.10.4 NO FLOW TESTS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 Deleted

3/4.11.1.2 Deleted

3/4.11.1.3 Deleted

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 Deleted

3/4.11.2.2 Deleted

3/4.11.2.3 Deleted

3/4.11.2.4 Deleted

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of oxygen below the specified values provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

This specification is not applicable to portions of the Waste Gas System Removed from service for maintenance, provided that the portions removed for maintenance are isolated from sources of hydrogen and purged of hydrogen to less than 4% by volume.

3/4.11.3 Deleted

3/4.11.4 Deleted

3/4.12 Deleted

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