ATTACHMENT 4
LICENSE AMENDMENT REQUEST
EXTENDED POWER UPRATE

TECHNICAL SPECIFICATIONS BASES MARKUPS

(For Information Only)

FLORIDA POWER & LIGHT
ST. LUCIE NUCLEAR POWER PLANT UNIT 2

This coversheet plus 22 pages

LIST OF PAGES

BASES FOR SECTION 2.0

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady-state peak linear heat rate below the level at which centerline fuel melting will occur. 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.

has

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 the ~~CE-1 or~~ ABB-NV correlation. The ~~CE-1 and~~ ABB-NV DNB correlations ~~have~~ been developed to predict the DNB heat flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined 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.

Specified Acceptable Fuel Design Limit for DNB (DNB-SAFDL)

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to the appropriate correlation limit for ~~DNB-SAFDL~~ in conjunction with the Extended Statistical Combination of Uncertainties (ESCU) or the revised Thermal Design Procedure (RTDP). This value is derived through a statistical combination of the system parameter probability distribution functions with the ~~CE-1 or~~ ABB-NV DNB correlation uncertainties. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions. /R4

2.1 SAFETY LIMITS (continued)

BASES (continued)

2.1.1 REACTOR CORE (continued)

vessel inlet

The curves of Figure 2.1-1 show conservative loci of points of THERMAL POWER, Reactor Coolant System pressure and ~~maximum cold leg~~ temperature with four Reactor Coolant Pumps operating for which the DNB-SAFDL is not violated based on the ABB-NV CHF correlation for the reference 1.55 Chopped Cosine Axial Shape and Design Limit F_r^T limit shown in Figure B 2.1-1. The dashed line is not a safety limit; however, operation above this line is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 107% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in Table 2.2-1. The area of safe transient condition is below and to the left of these lines. /R4

The conditions for the Thermal Margin Safety Limit curves in Figure 2.1-1 to be valid are shown on the figure.

DNB-SAFDL

The Thermal Margin/Low Pressure and Local Power Density Trip Systems, in conjunction with Limiting Conditions for Operation, the Variable Overpower Trip and the Power Dependent Insertion Limits, assure that the ~~Specified Acceptable Fuel Design Limits on DNB~~ and Fuel Centerline Melt are not exceeded during normal operation and design basis Anticipated Operational Occurrences. Specific verification of the DNB-SAFDL limit using an appropriate DNB correlation ensures that the reactor core safety limit is satisfied.

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 Coolant System components are designed to Section III, 1971 Edition including Addenda to the Summer, 1973, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System was hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS (continued)

BASES (continued)

2.2.1 REACTOR TRIP SETPOINTS (continued)

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated prior to or concurrently with a safety injection (SIAS). This also provides assurance that a reactor trip is initiated prior to or concurrently with an MSIS.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 626 psia is sufficiently below the full load operating point so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. ~~This setting was used with an uncertainty factor of 30 psi in the safety analyses.~~

Steam Generator Level-Low

The Steam Generator Level-Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to loss of the steam generator heat sink. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide sufficient time for any operator action to initiate auxiliary feedwater before reactor coolant system subcooling is lost. This trip also protects against violation of the specified acceptable fuel design limits (SAFDL) for DNBR, offsite dose and the loss of shutdown margin for asymmetric steam generator transients such as the opening of a main steam safety valve or atmospheric dump valve.

The trip setpoint is bounding relative to the accident and transient analyses which were performed using a lower, conservative trip setpoint. The trip setpoint and the methodology used to determine the trip setpoint, the as-found acceptance criteria band, and the as-left acceptance criteria are specified in the UFSAR. The two footnotes on the bottom of TS Table 2.2-1 are consistent with the two recommended notes provided in NRC's letter to the NEI Technical Setpoint Methods Task Force for Setpoint Allowables dated September 7, 2005.

3/4.1 REACTIVITY CONTROL SYSTEMS (continued)

BASES (continued)

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 makeup pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of the limit specified in the COLR after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions. This requirement can be met for a range of boric acid concentrations in the Boric Acid Makeup Tank (BAMT) and Refueling Water Tank (RWT). This range is bounded by ~~5350~~ 8750 gallons of ~~3.5~~ 3.1 weight percent (~~6119~~ 5420 ppm boron) from the BAMT and ~~16,000~~ 11,692 gallons of ~~1720~~ 1900 ppm borated water from the RWT to ~~8650~~ 7550 gallons of ~~2.5~~ 3.5 weight percent (~~4371~~ 6119 ppm boron) boric acid from BAMT and ~~12,000~~ 10,492 gallons of ~~1720~~ 1900 ppm borated water from the RWT. A minimum of ~~35,000~~ 33,000 gallons of ~~1720~~ 1900 ppm boron is required from the RWT if it is to be used to borate the RCS alone. This volume requirement, however, is expected to always be bounded by the ECCS RWT volume requirements of Specification 3.5.4.

With the RCS temperature below 200°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 ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

3/4.1 REACTIVITY CONTROL SYSTEMS (continued)

BASES (continued)

3/4.1.2 BORATION SYSTEMS (continued)

Temperature changes in the RCS impose reactivity changes by means of the moderator temperature coefficient. Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SDM. Small changes in RCS temperature are unavoidable and so long as the required SDM is maintained during these changes, any positive reactivity additions will be limited to acceptable levels. Introduction of temperature changes must be evaluated to ensure they do not result in a loss of required SDM.

can be satisfied by maintaining

The boron capability required below 200°F is based upon providing a SHUTDOWN MARGIN corresponding to its COLR limit after xenon decay and cooldown from 200°F to 140°F. This condition ~~requires~~ either 6750 gallons of
1900 ~~1720 ppm – 2100~~ ppm borated water from the refueling water tank or 3550 gallons of ~~2.5 to 3.5~~ weight percent boric acid solution from the boric acid makeup tanks.

3.1

The contained water volume limits includes 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.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 7.0 and 8.0 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.

Ensuring that the BAM pump discharge pressure is met satisfies the periodic surveillance requirement to detect gross degradation caused by impeller structural damage or other hydraulic component problems. Along with this requirement, Section XI of the ASME Code verifies the pump developed head at one point on the pump characteristic curve to verify both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the unit safety analysis. Surveillance Requirements are specified in the In-service Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and frequencies necessary to satisfy the requirements.

3/4.2 POWER DISTRIBUTION LIMITS (continued)

BASES (continued)

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and safety analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the appropriate correlation limit for DNB-SAFDL in conjunction with ESCU or RTDP methodology throughout each analyzed transient. /R2

These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. ~~However, the minimum RCS flow based on maximum analyzed steam generator tube plugging, is retained in the TS LCO~~. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient. /R2

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18-month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12-hour basis.

BASES FOR SECTION 3/4.4

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

the DNBR limit

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above ~~1.20~~ during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not 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 capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either shutdown cooling or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling loops be OPERABLE.

The operation of one reactor coolant pump or one shutdown cooling 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 reductions will, therefore, be within the capability of operator recognition and control.

If no coolant loops are in operation during shutdown operations, suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1.1 or 3.1.1.2 is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.2 SAFETY VALVES (continued)

Surveillance Requirements are specified in the Inservice Testing Program. Pressurizer code safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code, which provides the activities and the frequency necessary to satisfy the Surveillance Requirements. No additional requirements are specified. 3

The pressurizer code safety valve as-found setpoint is 2500 psia +/- ~~2~~% for OPERABILITY; however, the valves are reset to 2500 psia +/- 1% during the Surveillance to allow for drift. The LCO is expressed in units of psig for consistency with implementing procedures.

3/4.4.3 PRESSURIZER

A OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an Engineered Safety Features Actuation test signal concurrent with a loss of offsite power the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

Background (continued)

Specification 6.8.4.l has two parts to address the replacement SG and original SG designs. Specification 6.8.4.l.1. applies to the replacement SG design. TS 6.8.4.l.2 applies to the original SGs and contains requirements such as a sleeving repair method, alternate repair criteria and additional inspection requirements, which apply only to the original SG design and can be removed following SG replacement.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

Applicable Safety Analyses

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding a SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary-to-secondary leakage rate equal to the operational leakage rate limits in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes that contaminated secondary fluid is released via the main steam safety valves and/or atmospheric dump valves. The majority of the activity released to the atmosphere results from the tube rupture. activity in the d

two sources: 1)

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses the steam discharge to the atmosphere is based ~~on~~ the total primary-to-secondary leakage from all SGs of 0.5 gpm total and 0.25 gpm through any one SG ~~or is assumed to increase to 0.5 gpm total through all SGs and 0.25 gpm through any one SG~~ as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity ~~level of DOSE EQUIVALENT I-131~~ is assumed to be equal to the limits in LCO 3.4.8, "Reactor Coolant System Specific Activity~~."~~ For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2) and the requirements of 10 CFR 50.67 (Ref. 7).

, and 2) the pre-existing secondary side fluid inventory

/R7

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

," and the secondary coolant system activity is assumed to be equal to the limits in LCO 3.7.1.4, Plant Systems Activity."

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.8 SPECIFIC ACTIVITY

Replace with REACTOR COOLANT SYSTEM Insert 1

~~The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed 10 CFR 50.67 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 0.5 gpm total primary-to-secondary leakage through all SGs and 0.25 gpm through any one SG, and a loss of offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the St. Lucie site, such as site boundary location and meteorological conditions, were not considered in this evaluation.~~ /R7

~~The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.~~

~~Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture 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 correction 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.~~

INSERT 1
REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

The maximum allowable doses to an individual at the exclusion area boundary (EAB) distance for 2 hours following an accident, or at the low population zone (LPZ) outer boundary distance for the radiological release duration, are specified in 10 CFR 50.67 for design basis accidents using the alternative source term methodology and in Branch Technical Position 11-5 for the waste gas decay tank rupture accident. Dose limits to control room operators are given in 10 CFR 50.67 and in GDC 19.

The RCS specific activity LCO limits the allowable concentration of radionuclides in the reactor coolant to ensure that the dose consequences of limiting accidents do not exceed appropriate regulatory offsite and control room dose acceptance criteria. The LCO contains specific activity limits for both DOSE EQUIVALENT (DE) I-131 and DOSE EQUIVALENT (DE) XE-133.

The radiological dose assessments assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator tube leakage rate at the applicable Technical specification limit. The radiological dose assessments assume the specific activity of the secondary coolant is at its limit as specified in LCO 3.7.1.4, "Plant Systems - Activity."

The ACTIONS allow operation when DOSE EQUIVALENT I-131 is greater than 1.0 μCi/gram and less than 60 μCi/gram. The ACTIONS require sampling within four hours and every four hours following to establish a trend.

One surveillance requires the determination of the DE XE-133 specific activity as a measure of noble gas specific activity of the reactor coolant at least once per 7 days.

A second surveillance is performed to ensure that iodine specific activity remains within the LCO limit once per 14 days during normal operation and following rapid power changes when iodine spiking is more apt to occur. The frequency between two and six hours after a power change of greater than 15% RATED THERMAL POWER within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation.

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

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2 of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation. /R7

During heatup, the thermal gradients through the reactor vessel wall produce thermal stresses which are compressive at the reactor vessel inside surface and are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location when compared to the outside surface, the inside surface flaw may be more limiting. Consequently, for the heatup analysis both the inside and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and which are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface location. Since the neutron indication damage is also greatest at the inside surface location the inside surface flaw is the limiting location. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 50 degrees F per hour or cooldown rate of up to 100 degrees F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature ~~at 55 EFPY,~~ and t~~hey~~ include adjustments for pressure differences between the reactor vessel beltline and pressurizer instrument taps.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (continued)

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 will cause an increase in the RT_{NDT}. An adjusted reference temperature can be predicated using a) the initial RT_{NDT}, b) the fluence (E greater than 1 MeV), including appropriate adjustments for neutron attenuation and neutron energy spectrum variations through the wall thickness, c) the copper and nickel contents of the material, and d) the transition temperature shift as recommended by Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or other approved method. The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} ~~at 55 EFPY~~.

The actual shift in RT_{NDT} of the vessel materials will be benchmarked periodically during operation, by removing and evaluating, in accordance with 10 CFR 50 Appendix H and ASTM E185, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and the vessel inside radius are essentially identical, the measured transition temperature shift in RT_{NDT} for a set of material samples can be compared to the ~~predications~~ predictions of RT_{NDT} that were used for preparations of the pressure/temperature limits curves. If the measured delta RT_{NDT} values from the surveillance capsule are not conservatively within the measurement uncertainty of the prediction method, then heat up and cooldown curves must be re-evaluated.

The pressure-temperature limit lines shown on Figures 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements for Appendix G to 10 CFR 50.

The maximum RT_{NDT} all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 60°F. The Lowest Service Temperature limit line shown on Figures 3.4-2 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maxi-mum of 20% of the system's hydrostatic test pressure.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) (continued)

BASES (continued)

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and 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 safety injection tanks 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 hot leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

Insert 2 ECCS SUBSYSTEMS

TS 3.5.2, ACTION a.1. provides an allowed outage/action completion time (AOT) of up to 7 days from initial discovery of failure to meet the LCO provided the affected ECCS subsystem is inoperable only because its associated LPSI train is inoperable. This 7 day AOT is based on the findings of a deterministic and probabilistic safety analysis and is referred to as a “risk-informed” AOT extension. Entry into this ACTION requires that a risk assessment be performed in accordance with the Configuration Risk Management Program (CRMP) which is described in the Administrative Procedure (ADM-17.08) that implements the Maintenance Rule pursuant to 10 CFR 50.65.

In Mode 3 with RCS pressure < 1750 psia and in Mode 4, 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 trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provided this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0.

INSERT 2
ECCS SUBSYSTEMS

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

TS 3.5.2.c and 3.5.3 require that ECCS subsystem(s) have an independent OPERABLE flow path capable of automatically transferring suction to the containment on a Recirculation Actuation Signal. The containment sump is defined as the area of containment below the minimum flood level in the vicinity of the containment sump strainers. Therefore, the LCOs are satisfied when an independent OPERABLE flow path to the containment sump strainer is available.

TS 3.5.2.d requires that an ECCS subsystem(s) have an OPERABLE charging pump and associated flow path from the BAMT(s). Reference to TS 3.1.2.2 requires that the one charging pump flow path is from the BAMT(s) through the boric acid makeup pump(s). The charging pump flowpath is from the BAMT(s) through the gravity feed valves.

BASES FOR SECTION 3/4.6

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

limit the offsite radiation doses to within the limits of 10 CFR 50.67 during accident conditions

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 safety analyses. This restriction, in conjunction with the leakage rate limitation, will ~~ensure that the site boundary radiation doses are below the guidelines established for design basis accidents~~.

In accordance with Generic Letter 91-08, "Removal of ~~Component~~ Component Lists from Technical Specifications," the opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

3/4.6.1.2 CONTAINMENT LEAKAGE

43.48

The limitations on containment leakage rates ensure that total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a (~~41.8~~ psig) which results from the limiting design basis loss of coolant accident.

The surveillance testing for measuring leakage rates is performed in accordance with the Containment Leakage Rate Testing Program, and is consistent with the requirements of Appendix J of 10 CFR 50 Option B and Regulatory Guide 1.163 dated September, 1995, as modified by approved exemptions.

/R8

3/4.6 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.6.1 CONTAINMENT VESSEL (continued)

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

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 annulus atmosphere of 0.7 psi and (2) the containment peak pressure does not exceed the design pressure of 44 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 42.41 psig assuming the TS 3.6.1.4 ~~43.4 psig. The~~ limit of 0.4 psig for initial positive containment pressure . This ~~will limit the total pressure to 43.99 psig which~~ is less than the design pressure of 44 psig and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment vessel average air temperature ensures that the containment peak temperature does not exceed the containment vessel design temperature of 264°F during steam line break conditions and is consistent with the safety analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

The limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of ~~41.8~~ 43.48 psig in the event of the limiting design basis loss of coolant accident. A visual inspection in accordance with the Containment Leakage Rate Testing Program is sufficient to demonstrate this capability.

BASES FOR SECTION 3/4.7

adequate to maintain secondary side pressure below 110% of the design value after

3/4.7 PLANT SYSTEMS

BASES

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% (1100 psia) of its design pressure of 1000 psia during the most severe anticipated system operational transient. 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 Vessel Code, 1971 Edition, and ASME Code for Pumps and Valves, Class II. The total relieving capacity for all valves on all of the steam lines is 12.49×10^6 lbs/hr ~~which is 103.8% of the total secondary steam flow of 12.03 x 10^6 lbs/hr at 100% RATED THERMAL POWER~~. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip set-point reductions are derived on the following bases:

For two loop operation:

$$SP = \left[\frac{(X)-(Y)(V)}{X} \times (107.0)\right] - 0.9$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

3/4.7 PLANT SYSTEMS (continued)

BASES (continued)

3/4.7.1 TURBINE CYCLE (continued)

3/4.7.1.1 SAFETY VALVES (continued)

107.0 = Power Level-High Trip Setpoint for two loop operation

0.9 = Equipment processing uncertainty

X = Total relieving capacity of all safety valves per steam line in lbs/hour (6.247×10^6 lbs/hr)

Y = Maximum relieving capacity of any one safety valve in lbs/hour (7.74×10^5lbs/hr)

Surveillance Requirement 4.7.1.1 verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The MSSV setpoints are 1000 psia ~~+1/-3%~~ +/-3% (4 valves each header) and 1040 psia ~~+1/-3%~~ +2%/-3% (4 valves each header) for OPERABILITY; however, the valves are reset to 1000 psia +/- 1% and 1040 psia +/- 1%, respectively, during the Surveillance to allow for drift. The LCO is expressed in units of psig for consistency with implementing procedures.

The provisions for Specification 3.0.4 do not apply. This allows entry into and operation in MODE 3 prior to performing the Surveillance Requirement so that the MSSVs may be tested under hot conditions.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system pumps ensures that the Reactor Coolant System can be cooled down to less than ~~350~~ 325°F from normal operating conditions in the event of a total loss-of-offsite power.

Each electric-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 320 gpm at a pressure of 1000 psia to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 500 gpm at a pressure of 1000 psia to the entrance of the steam generators. This capacity is sufficient to ensure adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than ~~350~~ 325°F when the shutdown cooling system may be placed into operation.

3/4.7 PLANT SYSTEMS (continued)

BASES (continued)

3/4.7.1 TURBINE CYCLE (continued)

3/4.7.1.3 CONDENSATE STORAGE TANKS

The OPERABILITY of the condensate storage tank with the minimum water volume of 307,000 gallons required by the LCO ensures that sufficient water is available to maintain the Unit 2 RCS at HOT STANDBY conditions for 4 hours followed by an orderly cooldown to the shutdown cooling entry temperature (~~350~~ 325°F). The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The actual water requirements are ~~149,600~~ 154,000 gallons for Unit 2 and ~~125,000~~ 130,500 gallons for Unit 1. ~~Included in the~~ The required volumes of water ~~are~~ also includes the tank unusable volume of ~~9400~~ 9203 gallons and ~~a conservative~~ an allowance for instrument error of ~~21,400~~ 4230 gallons.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will comply with the dose criterion provided in 10 CFR 50.67 in the event of a steam line rupture. The 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 and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the safety analyses. /R4

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 blow down 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 steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements is consistent with the assumptions used in the safety analyses.

The specified 6.75 second full closure time represents the addition of the maximum allowable instrument response time of 1.15 seconds and the maximum allowable valve stroke time of 5.6 seconds. These maximum allowable values should not be exceeded because they represent the design basis values for the plant.

BASES FOR SECTION 3/4.11

3/4.11 RADIOACTIVE EFFLUENTS

BASES

Pages B 3/4 11-2 through B 3/4 11-3 (Amendment No. 61) have been deleted from the Technical Specifications. The next page is B 3/4 11-4.

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 hydrogen and oxygen below their flammability limits 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.

3/4.11.2.6 GAS STORAGE TANKS

gaseous radioactive waste inventory in a

effective dose equivalent

Restricting the ~~quantity of radioactivity contained in each~~ gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total ~~body exposure~~ to an individual at the nearest exclusion area boundary will not exceed ~~0.5~~ rem. This is consistent with ~~Standard Review Plan 15.7.1, "Waste Gas System Failure."~~

0.1

Branch Technical Position 11-5, "Postulated Radioactive Release Due to Waste Gas System Leak or Failure," of Standard Review Plan Chapter 11, Radioactive Waste Management," of NUREG-0800. The waste gas decay tank inventory of noble gases required to generate an exclusion area boundary dose of 0.1 rem is the basis for the limit of 202,500 dose equivalent curies Xe-133, and is derived based on the definition given in Technical Specification Task Force (TSTF-490), "Deletion of E bar Definition and Revision to RCS Specific Activity Tech Spec."