ATTACHMENT 4
LICENSE AMENDMENT REQUEST
EXTENDED POWER UPRATE

TECHNICAL SPECIFICATIONS BASES MARKUPS

(For Information Only)

FLORIDA POWER AND LIGHT
ST. LUCIE NUCLEAR PLANT UNIT 1

This coversheet plus 24 pages

LIST OF PAGES

<u>Technical Specifications Bases</u>

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.

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 measured parameter during operation and therefore THERMAL POWER, Reactor Coolant Temperature and Pressure have been related to DNB using a DNB correlation developed to predict the Critical Heat Flux (CHF) for DNB. The CHF is the heat flux at a particular core location that would cause DNB. The ratio of the CHF to the actual local heat flux at a particular core location is called the DNB Ratio (DNBR) and is indicative of the margin to DNB.

The minimum allowed value of the DNBR during steady state operation, normal operational transients, and anticipated transients is the DNBR limit from the appropriate DNB correlation. The DNBR limit corresponds to a 95% probability at a 95% confidence level that DNB will not occur at a particular core location, providing appropriate margin to DNB for all operating conditions. In a core with fuel assemblies of different designs (mixed core), there may be more than one DNB correlation and associated DNBR limit that defines DNB for the core.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure, and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the DNBR limit corresponding to the ~~XNB~~ HTP DNB correlation is not violated for the following conditions: /R1

~~NOTE: These curves remain bounding for the use of HTP DNB correlation with respect to the violation of DNBR limit.~~ /R1

2.1 SAFETY LIMITS (continued)

BASES (continued)

2.1.1 REACTOR CORE (continued)

1. reactor coolant inlet temperatures less than or equal to 580°F,

2. THERMAL POWER less than or equal to 112%,

3. reactor coolant vessel flow ~~of 365,000 gpm~~ within the limits specified in COLR Table 3.2-1, and

4. the axial power shape shown on Figure B2.1-1.

The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F 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 112% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in Table 2.2-1. The area of safe operation is below and to the left of these lines.

The reactor protective system in combination with the Limiting Conditions for Operation is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than the DNBR limit and preclude the existence of flow instabilities. Specific verification of the DNBR limit with 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 pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves and fittings are designed to ANSI B 31.7, Class I which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is 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)

Steam Generator Water Level-Low

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the design pressure of the reactor coolant system will not be exceeded due to loss of steam generator heat sink. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to provide sufficient time for any operator action to initiate auxiliary feedwater before reactor coolant system subcooling is lost.

Local Power Density-High

The local Power Density-High trip, functioning from AXIAL SHAPE INDEX monitoring, is provided to ensure that the peak local density in the fuel which corresponds to fuel centerline melting will not occur as a consequence of axial power maldistributions. A reactor trip is initiated whenever the AXIAL SHAPE INDEX exceeds the allowable limits of Figure 2.2-2. The AXIAL SHAPE INDEX is calculated from the upper and lower ex-core neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level with the allowed CEA group position being inferred from the THERMAL POWER level. The trip is automatically bypassed below 15 percent power.

The maximum AZIMUTHAL POWER TILT and maximum CEA misalignment permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with the Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

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 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 all operating conditions corresponding to the
1900 requirements of Specification 3.1.1.2 after xenon decay and cooldown to
200°F. The maximum boration capability requirement occurs at EOL from
full power equilibrium xenon conditions. This requirement can be met for
1900 a range of boric acid concentrations in the Boric Acid Makeup Tanks
6800 (BAMTs) and Refueling Water Tank (RWT). This range is bounded by 5245
~~5400~~ gallons of 3.5 weight percent (6119 ppm boron) boric acid from the
BAMTs and 17,000 gallons of ~~1720~~ ppm borated water from the RWT to
3.0 8700 gallons of ~~2.5~~ weight percent (~~4371~~ ppm boron) boric acid from the
BAMTs and 13,000 gallons of ~~1720~~ ppm borated water from the RWT.
A minimum of 45,000 gallons of ~~1720~~ ppm boron is required from the
RWT if it is to be used to borate the RCS alone.
1900

The requirements for a minimum contained volume of 401,800 gallons of borated water in the refueling water tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified here ~~too~~.
as well

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 change 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.

The boron addition capability after the plant has been placed in MODES 5 and 6 requires either 3650 gallons of ~~2.5~~ 3.0 to 3.5 weight percent boric acid solution (~~4371~~ 5245 to 6119 ppm boron) from the boric acid tanks or 11,900 gallons of ~~1720~~ 1900 ppm borated water from the refueling water tank to makeup for contraction of the primary coolant that could occur if the temperature is lowered from 200°F to 140°F.

The restrictions associated with the establishing of the flow path from the RWT to the RCS via a single HPSI pump provide assurance that 10 CFR 50 Appendix G pressure/temperature limits will not be exceeded in the case of any inadvertent pressure transient due to a mass addition to the RCS. If RCS pressure boundary integrity does not exist as defined in Specification 1.16, these restrictions are not required. Additionally, a limit on the maximum number of operable HPSI pumps is not necessary when the pressurizer manway cover or the reactor vessel head is removed.

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 Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and frequencies necessary to satisfy the requirements. /R2

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 accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than or equal to the DNBR limit throughout each analyzed transient.

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.

The limits are cycle-specific and have been relocated to the COLR.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

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

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

activity in the

d

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses the steam discharge to the atmosphere is based on the total primary-to-secondary leakage from all SGs of 1 gpm and 0.5 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)~~, 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits)~~.

two sources :1)

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

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

and 10 CFR 50.67 (Ref. 7).

Limiting Condition for Operation (LCO)

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

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

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.l, “Steam Generator 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.

/R3

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduce the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limit time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

Replace Specific Activity Bases with REACTOR COOLANT SYSTEM Insert 1

3/4.4.8 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 and a concurrent 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.~~

~~The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/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.~~

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)

~~3/4.4.8 SPECIFIC ACTIVITY (continued)~~

~~Reducing T_{avg} to < 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.~~

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 5.2.1 of the FSAR. 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.

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 surface 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 than at the outside surface location, the inside surface flaw may be more limiting. Consequently, for the heatup analysis, both the inside surface 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 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.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.9 PRESSURIZER/TEMPERATURE LIMITS (continued)

Since neutron irradiation damage is also greater at the inside surface, the inside surface flaw location is the limiting location during cooldown. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

The heatup and cooldown limit curves (Figures 3.4-2a and 3.4-2b) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for the heatup rate of up to 70 ~~50~~°F/hr and for any cooldown rate of up to 100°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the applicable service period.

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>1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature can be calculated based upon the fluence. The heatup and cooldown limit curves shown on Figures 3.4-2a and 3.4-2b include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for pressure differences between the reactor vessel beltline and pressurizer instrument taps.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-82, reactor vessel material surveillance specimens installed near the inside wall of the reactor vessel in the core area. The capsules are scheduled for removal at times that correspond to key accumulated fluence levels within the vessel through the end of life. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, measured ΔRT_{NDT} for surveillance samples can be applied with confidence to the corresponding material in the reactor vessel wall. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figures 3.4-2a and 3.4-2b 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.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.9 PRESSURIZER/TEMPERATURE LIMITS (continued)

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been established to be ~~90~~°F. The Lowest Service Temperature limit line shown on Figures 3.4-2a and 3.4-2b is based upon this RT_{NDT} since Article NB-2332 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 maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

58

This Lowest Service Temperature value of 165°F also includes an additional 7°F to account for temperature measurement uncertainty.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection program for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. This program is in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a (g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1971 Edition and Addenda through Winter 1972.

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.11 DELETED

3/4.4.12 PORV BLOCK VALVES

The opening of the Power Operating Relief Valves fulfills no safety related function. The electronic controls of the PORVs must be maintained OPERABLE to ensure satisfaction of Specifications 3.4.12 and 3.4.13. Since it is impractical and undesirable to actually open the PORVs to demonstrate reclosing, it becomes necessary to verify operability of the PORV Block Valves to ensure the capability to isolate a malfunctioning PORV.

3/4.4.13 POWER OPERATED RELIEF VALVES and
3/4.4.14 REACTOR COOLANT PUMP - STARTING

The low temperature overpressure protection system (LTOP) is designed to prevent RCS overpressurization above the 10 CFR 50 Appendix G operating limit curves (Figures 3.4-2a and 3.4-2b) at RCS temperatures at or below ~~304~~ 300 °F ~~during heatup and 281°F during cooldown~~. The LTOP system is based on the use of the pressurizer power-operated relief valves (PORVs) and the implementation of administrative and operational controls.

The PORVs aligned to the RCS with the low pressure setpoints of 350 and 530 psia, restrictions on RCP starts, limitations on heatup and cooldown rates, and disabling of non-essential components provide assurance that Appendix G P/T limits will not be exceeded during normal operation or design basis overpressurization events due to mass or energy addition to the RCS. The LTOP system APPLICABILITY, ACTIONS, and SURVEILLANCE REQUIREMENTS are consistent with the resolution of Generic Issue 94, “Additional Low-Temperature Overpressure Protection for Light-Water Reactors,” pursuant to Generic Letter 90-06.

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 cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

TS 3.5.2.c and 3.5.3.a require that ECCS subsystem(s) have an independent OPERABLE flow path capable of automatically transferring suction to the containment sump 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. /R2

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.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained.

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

BASES FOR SECTION 3/4.6

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 CONTAINMENT VESSEL

3/4.6.1.1 CONTAINMENT VESSEL INTEGRITY

CONTAINMENT VESSEL 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 Part 100 during accident conditions.

In accordance with Generic Letter 91-08, "Removal of 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 this action will prevent the release of radioactivity outside the containment.

3/4.6.1.2 CONTAINMENT LEAKAGE

42.77

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 (~~39.6~~ 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 Rev. 0, as modified by approved exemptions.

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 CONTAINMENT SYSTEMS (continued)

<u>BASES</u> (continued)

3/4.6.1 CONTAINMENT VESSEL (continued)

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structural is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.70 psi and 2) the containment peak pressure does not exceed the design pressure of 44 psig during steam line break accident conditions. 0.5

43.08

The maximum peak pressure obtained from a steam line break accident is ~~41.6~~ psig. The limit of ~~2.4~~ psig for initial positive containment pressure will limit the total pressure to 44.0 psig which is the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment air temperature ensures that the containment vessel temperature does not exceed the design temperature of 264°F during LOCA conditions. The containment temperature limit is consistent with the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY 42.77

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 ~~39.6~~ 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.

3/4.6 CONTAINMENT SYSTEMS (continued)

BASES (continued)

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS (continued)

3/4.6.2.1 CONTAINMENT SPRAY AND COOLING SYSTEMS (continued)

Ensuring that the containment spray 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 Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and frequencies necessary to satisfy the requirements.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

7.0

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 a pH value of between ~~8.5~~ and 11.0 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. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

3/4.6.2.3 DELETED

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. This includes the containment purge inlet and outlet valves.

BASES FOR SECTION 3/4.7

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% 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 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 is12.38 x 10^6 lbs/hr ~~which is 102.8 percent the total secondary steam flow of 12.04 x 10^6 lbs/hr at 100% RATED THERMAL POWER~~. A minimum of 2 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 setpoint reductions are derived on the following bases:

‘For two loop operation

$$SP = \frac{(X)-(Y)(V)}{X} \text{ x } (106.5)$$

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)

<u>**BASES**</u> (continued)

3/4.7.1 TURBINE CYCLE (continued)

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

X = Total relieving capacity of all safety valves per steam line in lbs/hour (6.192×10^6 lbs/hr.)

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

+2/-3%

+/- 3%

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%~~ (4 valves each header) and 1040 psia ~~+1/-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 of Specification 3.0.4 do not apply. This allows entry into and operation in MODE 3 prior to performing the Surveillance Requirements so that the MSSVs may be tested under hot conditions.

3/4.7.1.2 AUXILIARY FEEDWATER PUMPS

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

Any two of the three auxiliary feedwater pumps have the required capacity to provide sufficient feedwater flow to remove reactor decay heat and reduce the RCS temperature to 325°F where the shutdown cooling system may be placed into operation for continued cooldown.

3/4.7.1.3 CONDENSATE STORAGE TANKS

to maintain HOT STANDBY for 1 hour and then

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available ~~for~~ cooldown of the Reactor Coolant System to less than 325°F in the event of a ~~total~~ loss of off-site power. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to atmosphere.

The minimum usable volume to satisfy the criteria stated above is 130,500 gallons, which is ensured by the LCO for the CST volume of 153,400 gallons.

3/4.7 PLANT SYSTEMS (continued)

<u>BASES</u> (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 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°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 gallons for Unit 2 and ~~125,000~~ 130,500 gallons for Unit 1. Included in the required volumes of water are 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.

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DELETED

~~3/4.9 REFUELING OPERATIONS (continued)~~

~~**BASES** (continued)~~

~~3/4.9.14 DECAY TIME – STORAGE POOL~~

~~The minimum requirements for decay of the irradiated fuel assemblies in the entire spent fuel storage pool prior to movement of the spent fuel cask into the fuel cask compartment ensure that sufficient time has elapsed to allow radioactive decay of the fission products. The decay time of 1180 hours is based upon one-third of a core placed in the spent fuel pool each year during refueling until the pool is filled. The decay time of 1490 hours is based upon one-third of a core being placed in the spent fuel pool each year during refueling following which an entire core is placed in the pool to fill it. The cask drop analysis assumes that all of the irradiated fuel in the filled pool (7 2/3 cores) is ruptured and follows Regulatory Guide 1.25 methodology, except that a Radial Peaking Factor of 1.0 is applied to all irradiated assemblies.~~

TS 3/4.9.14 Decay Time - Storage Pool has been deleted.

BASES FOR SECTION 3/4.11

3/4.11 RADIOACTIVE EFFLUENTS

<u>BASES</u>

Pages B 3/4 11-2 through B 3/4 11-3 (Amendment No. 123) 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 con-trolled 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

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

effective dose equivalent

~~nearest~~ exclusion area boundary will not exceed 0.~~5~~ rem. This is

1

consistent with ~~Standard Review Plan 15.7.1, "Waste Gas System Failure."~~

Branch Technical Position 11-5, "Postulated Radioactive Releases 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 source term 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."