SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
TECHNICAL SPECIFICATION CHANGE REQUEST
STEAM GENERATOR REPLACEMENT

Cover Letter

Enclosure 1 - BASIS FOR PROPOSED CHANGES

Enclosure 2 - NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (10 CFR 50.92)

Enclosure 3 - ENVIRONMENTAL EVALUATION (10 CFR 51.22)

Enclosure 4 - PAGE CHANGE INSTRUCTIONS

Enclosure 5 - PROPOSED TECHNICAL SPECIFICATION PAGE CHANGES

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
TECHNICAL SPECIFICATION CHANGE REQUEST
STEAM GENERATOR REPLACEMENT

BASIS FOR PROPOSED CHANGES

(26 pages total)

BASIS FOR PROPOSED CHANGE

INTRODUCTION

Carolina Power & Light (CP&L) has completed a comprehensive engineering program to replace the original Westinghouse Model D4 (OSGs) with Westinghouse Model Delta 75 replacement steam generators (RSGs) at the Harris Nuclear Plant (HNP). This license amendment application requests approval for steam generator replacement and restoration of the original reactor coolant average temperature (T_{avg}) of 588.8°F. Results of the analyses and evaluations support operation at an uprated reactor core power level of 2900 MWt; however, uprated power operation is not being requested by this license amendment application.

The analyses and evaluations address plant operation over a range of T_{avg}, up to the original HNP T_{avg} of 588.8°F, and an all-Siemens fuel core design that supports operation at both the uprated and current power levels. A single set of analyses that encompasses steam generator replacement (SGR), a range of operating conditions, and the power uprate option was performed to provide conservative results that bound operation at the current power level following replacement. Technical Specifications (TS) changes needed for support of operations with the RSGs at the existing power level and T_{avg} of 588.8°F are summarized in Table 1. Since the Technical Specification changes are generated from a single set of analyses that considered bounding conditions, the changes do not necessarily result from SGR alone but are appropriate for operation with the RSGs at the current reactor core power level of 2775 MWt.

The engineering review program that considered SGR and the power uprate option, hereinafter referred to as the Project, was performed by a team managed by CP&L that includes Siemens Power Corporation (SPC), Raytheon Nuclear, Inc. (now Washington Group Inc, or WGI), Westinghouse Electric Company, and others. CP&L provided and controlled the design basis inputs used in the evaluations, established the interface between team members, and conducted formal owner's review, approval, and integration of results. The Project includes evaluation of impacts to NSSS and BOP systems, identification of required modifications, and re-analysis or evaluation of postulated design transients. The results of the analyses and evaluations presented in the NSSS and BOP License Reports (Enclosures 6 and 7, respectively) demonstrate applicable acceptance criteria continue to be met.

In order to support both SGR and the power uprate option, the Project review is consistent with the methodology established in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant." This methodology is based on utilizing CP&L-approved analysis input assumptions/parameter values, approved analytical techniques (e.g., methodologies and computer codes), and currently applicable licensing criteria and standards. The assessments of plant systems, components and structures were performed with respect to operating conditions at the uprated power level and the results evaluated for acceptability at current power. Where appropriate, additional analyses were performed assuming current power level to ensure that the uprated results are bounding for both power levels.

Analytical Techniques

The Project analytical techniques (methodology and tools) and engineering efforts are discussed in detail in Enclosures 6 and 7. The analytical techniques are consistent with current HNP licensing bases except where an approved updated analytical model was considered more appropriate. For example, the license basis analysis for main steam line break (MSLB) mass and energy (M&E) releases was updated by using WCAP 10961 (approved in Farley Nuclear Plant submittal) and WCAPS8822-S1-P-A and 8822-S2-P-A . The steam generator tube rupture (SGTR) radiological consequences analysis was updated to use the latest resolution of the steam generator tube uncovery issue as documented in WCAP-13247 (also approved for Farley). The application of leak-before-break (LBB) criteria to reactor coolant piping was previously documented for HNP in WCAP-10699 and approved by the NRC. In 1996, the LBB analysis was updated for the replacement steam generator using the latest criteria. Analysis results are documented in WCAP-14549 (described in Enclosure 6) and were re-evaluated as acceptable for Project-specified conditions.

Events and transients described in the FSAR, Chapter 15 were either re-analyzed or evaluated as bounded by another analyzed event. Major re-analysis efforts are identified below and discussed in more detail in Enclosure 6 and the basis discussions that follow.

The increase in RCS volume resulting from the larger RSG tube volume dictated comprehensive re-analyses, including the loss of coolant accident (LOCA). The LOCA re-analysis verified existing Technical Specification peaking limits and axial dependent power peaking limit (K(z)) curve as well as the adequacy of the Emergency Core Cooling System (ECCS) by demonstrating that regulatory requirements are met. The conditions assumed for LOCA re-analysis include the RSG with up to 3% tube plugging and rated thermal reactor power of 2900 MWt. The calculated peak fuel cladding temperature (PCT) is increased over the current HNP license basis value but remains within the 10CFR50.46(b) acceptance criteria limit of 2,200°F. The Project analysis results for both large break and small break LOCA are consistent with the HNP license basis and meet the other 10CFR50.46(b) acceptance criteria.

The feedwater line break (FWLB) event was re-analyzed to incorporate Project-specified conditions and to reflect the change to the upper SG feedwater nozzle location associated with the RSG feedring design. The containment internal structure (SG subcompartment walls and internal structure) was evaluated for a FWLB at the SG nozzle and stresses are within design allowables.

The SGTR event was re-analyzed to consider impact of several RSG design differences, including a larger heat transfer area (UA), smaller secondary volume, and smaller tube rupture area.

Core design

The core design upon which the safety analyses and other evaluations are based is a CP&L developed "Equilibrium Fuel Cycle," representative of post-replacement and uprated power core loadings and energy requirements. The cycle-specific enrichment and burnable absorber locations may vary within generically accepted limits for Siemens high thermal performance (HTP) fuel designs, and the number and placement of new fuel assemblies in the core may change, but these variations are within the operating experience and normal variations expected for reload fuel and core designs.

The core design was validated by Siemens to meet general core acceptance criteria and the loading pattern used to develop necessary neutronics input to the FSAR safety analyses and to develop bounding core characteristics for source term determination. The fuel mechanical, thermal/hydraulic and nuclear design characteristics have not changed. An appropriate core design for operation following SG replacement is demonstrated by the results of the safety analyses, since applicable acceptance criteria continue to be met.

The proposed Technical Specifications changes (Table 1) are supported by overall program results. The results of analyses and evaluations are consistent with and continue to comply with the current HNP licensing basis acceptance criteria. The results of analyses and evaluations performed at the uprated reactor core power of 2900 MWt bound operation at the current reactor core power of 2775 MWt.

BASIS FOR PROPOSED CHANGE

TABLE 1

LIST OF PROPOSED TECHNICAL SPECIFICATION CHANGES

TS No.	Page No.	Title
Group I		**Analysis Limits for RCS & Containment**
5.4.2	5-6	RCS Volume at T_{avg} = 588.8°F
6.8.4.k B 3/4.6.1.4 B 3/4.6.1.6	6-19c B 3/4 6-1a B 3/4 6-2	Containment Leak Test Program Containment Systems
4.4.1.2.2 4.4.1.3.2 3.4.1.4.1.b 4.4.1.4.1.1 B 3/4.4.1	3/4 4-3 3/4 4-5 3/4 4-6 B 3/4 4-1	RCS - Hot Standby RCS - Hot Shutdown RCS - Cold Shutdown RCS and Coolant Circulation
3.4.1.2 3.10.4 4.10.4.3	3/4 4-2 3/4 10-4	RCS - Hot Standby Special Test Exceptions RC Loops
3.1.2.5 B 3/4.1.2	3/4 1-11 B 3/4 1-3	Borated Water Source - Shutdown
Group II		**Core Limits and RTS/ESFAS**
Fig. 2.1-1 B 2.1.1 3.2.5.a B 3/4.2.5 6.9.1.6	2-2 B 2-1,6 3/4 2-14 B 3/4 2-6 6-24a,b,c	Core Safety Limit Line Plot DNB Parameters Core Operating Limits Report
Table 2.2-1 Table 3.3-4	2-4,5,7-10 3/4 3-28 3/4 31-34	RTS Instrument Trip Setpoints ESFAS Instrument Trip Setpoints
Group III		**Dose and Specific Activity**
3.4.8 Fig. 3.4-1 Table 4.4-4 B 3/4.4.8 1.11	3/4 4-29 3/4 4-30 3/4 4-31 B 3/4 4-5 1-2a	Specific Activity Definitions – Dose Equivalent I-131
Group IV		**Plant Systems & Other**
Table 3.7-1	3/4 7-2	Max Power Range Neutron Flux High Setpoint with Inoperable MSSVs
B 3/4.7.1.3	B 3/4 7-2	Condensate Storage Tank
3/4.4.5 B 3/4.4.5	3/4 4-13 to 17 3/4 4-19 to 20a B3/4 4-3	SG Surveillance Requirements
B 3/4.7.2	B3/4 7-2	SG Pressure/Temperature Limitation
B 3/4.7.1.2	B3/4 7-2	Auxiliary Feedwater System

BASIS FOR PROPOSED CHANGE

The background and bases for the proposed Technical Specification changes are discussed below in the following four groups:

I. Analysis Limits for RCS and Containment
II. Core Safety Limits and Reactor Trip System/Engineered Safety Feature Actuation System Trip Setpoints and Associated Values
III. Specific Activity and Dose
IV. Plant Systems and Other

Group I Analysis Limits for RCS & Containment

TS #	Title
5.4.2	RCS Volume
6.8.4.k B 3/4.6.1.4 B 3/4.6.1.6	Containment Leak Test Program Containment Systems
4.4.1.2.2 4.4.1.3.2 3.4.1.4.1.b 4.4.1.4.1.1 B 3/4.4.1	Reactor Coolant System – Hot Standby Reactor Coolant System – Hot Shutdown Reactor Coolant System – Cold Shutdown RCS and Coolant Circulation
3.4.1.2 3.10.4 4.10.4.3	Reactor Coolant System – Hot Standby Special Test Exceptions RC Loops
3.1.2.5 B 3/4.1.2	Borated Water Source - Shutdown

BACKGROUND

RCS Volume

The Harris Nuclear Plant (HNP) Technical Specification (TS) 5.4.2 (Reactor Coolant System Volume) currently specifies a total water and steam volume of 9410 ± 100 cubic feet at a nominal T_{avg} of 580.8°F. . As a result of design differences between the RSG and OSG, the RCS volume is increased following SGR. The increase is due to differences in the RSG vs OSG tube bundle, namely, the number and length of u-tubes is greater for the RSGs. Total RCS volume information is used as a design input to several of the transient and accident analyses described in Chapter 15 of the HNP FSAR. For these analyses, uncertainties and conservatisms are applied to the best estimate value in accordance with approved methodologies.

RHR Cooldown

The residual heat removal (RHR) system removes heat from the RCS during normal cooldown and accident conditions. RHR cooldown is initiated when RCS temperature has been reduced to approximately 350°F and continues operating to achieve cold shutdown (200°F) or refueling (140°F) conditions. Normal RHR cooldown utilizes both safety trains to minimize cooldown

times. Single train RHR cooldown satisfies the Standard Review Plan by meeting single failure criteria and by cooldown to 200°F within a reasonable period of time.

Peak Containment Pressure

The HNP containment leakage rate testing program is required by 10 CFR 50.54 (o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. These requirements define P_a as the calculated peak internal pressure related to the design basis accident. Analyses determine P_a using instrument uncertainties and conservative values for appropriate parameters, such as initial positive containment pressure, to ensure that the design pressure (45 psig) is not exceeded during design basis events. Thus, containment integrity is maintained and leakage rates are acceptably low during postulated accidents.

Existing Technical Specification 6.8.4.k (Containment Leakage Rate Testing Program) reflects calculated peak containment internal pressures of 38.4 psig due to a LOCA and 41.2 psig due to the MSLB accident. The larger of the two is conservatively assumed to be the value of P_a for the purpose of containment testing in accordance with the Technical Specification. The Bases statements for Technical Specification 3/4.6.1.4 (Internal Pressure) and Technical Specification 3/4.6.1.6 (Containment Structural Integrity) reflect the MSLB peak pressure value.

SG Water Level/RC Loop Operability

Technical Specification 3/4.4.1 specifies minimum water levels as limiting conditions or surveillance requirements to support an adequate heat sink for the removal of decay heat. The water level values include allowances for instrument channel uncertainty and process measurement effects. Technical Specification 3/4.10.4 provides a special test exception to mode 3 loop operability requirements during the performance of hot rod drop time measurements.

Borated Water Source

Technical Specification 3.1.2.5 specifies the minimum operable borated water source requirements during shutdown. The specified minimum usable volume for the boric acid tank is sufficient to provide the required shutdown margin after xenon decay and cooldown from 200 to 140°F.

PROPOSED CHANGES

CP&L proposes to modify HNP Technical Specification 5.4.2 to reflect the increase in RCS volume due to the larger tube bundle of the RSGs. The proposed value of "approximately 10,300 cubic feet" is specified at a nominal T_{avg} of 588.8°F since T_{avg} will be restored to the original HNP design basis value of 588.8°F.

Technical Specification 6.8.4.k is modified to indicate peak calculated pressures for the design basis MSLB and LOCA of 41.3 psig and 41.8 psig (respectively). The LOCA value will be used

for peak pressure (P_a) in Technical Specification Bases 3/4.6.1.4 and 3/4.6.1.6 to reflect the results of accident analyses performed for SG replacement and restoration of T_{avg} to 588.8 °F.

Technical Specification 3/4.4.1 Surveillance Requirements (SR) 4.4.1.2.2, 4.4.1.3.2, 4.4.1.4.1.1 and Limiting Condition for Operation (LCO) 3.4.1.4.1.b and associated Bases are modified to provide a revised SG minimum level limit of 30% narrow range (NR) for Modes 3, 4, and 5 and to specify that a SG minimum water level limit of 74% wide range (WR) shall be used for Modes 4 and 5, unless inoperable, to support limiting conditions for operation and surveillance requirements for the RSGs.

Technical Specification 3/4.10.4 is modified to delete an unnecessary test exception for the performance of hot rod drop time measurements.

Technical Specification 3.1.2.5 and Bases are revised to specify a minimum contained borated water volume for the boric acid tank of 7,150 gallons and associated indicated level of 23%.

BASIS

RCS Volume

The comprehensive engineering review effort performed in support of SGR includes evaluation or re-analysis of license basis transients and accidents. The review results demonstrate that applicable acceptance criteria, including DNB, RCS pressure and temperature, peak cladding temperature, containment pressure and temperature, and dose limits continue to be met. In addition, plant systems/parameters with expected impact from the increase in mass and heat energy represented by the RSGs primary volume increase were evaluated. These include residual heat removal (RHR) cooldown time and peak containment pressure. The evaluations are discussed below.

RHR Cooldown

The RSGs increase the times associated with plant cooldown. The dual train RHR cooldown evaluation (normal cooldown) initiates RHR at an RCS temperature of 350°F and ends at the refueling mode temperature, 140°F. No time period is specified in the regulatory requirement for normal cooldown (557°F to 140°F). For the single train RHR cooldown evaluation, RHR is initiated at an RCS temperature of 350°F and cooldown is complete at the cold shutdown temperature, 200°F. The time period required for single train cooldown (557°F to 200°F) is "a reasonable period of time." The HNP Safety Evaluation Report (SER) reported a time of 18 hours for single train RHR cooldown, although the stated time is more compatible with existing dual train RHR cooldown.

The Project dual train cooldown analysis results in a time of approximately 22 hours. Conservative assumptions are included for appropriate input values such as spent fuel pool heat load and service water temperature. For single train cooldown, cases with and without the availability of offsite power were evaluated. The limiting case, with offsite power, determined a

cooldown time of approximately 40 hours for operation at 2775 MWt and 44 hours for operation at reactor core power of 2900 MWt. This satisfies the Standard Review Plan requirement that single train cooldown should be accomplished in a reasonable period of time.

Peak Containment Pressure

The LOCA and MSLB design basis accidents have been re-analyzed in support of SGR, restoration of T_{avg} to 588.8°F, and the power uprate option. These postulated accidents involve a relatively large, uncontrolled amounts of mass and energy (M&E) release into containment, producing maximum internal pressure levels. Calculated M&E releases are used as input to the containment integrity analysis to demonstrate the acceptability of the containment safeguards systems to mitigate the consequences of such postulated events. The calculated peak pressures must be less than the containment design pressure to ensure containment integrity. In addition to limiting peak pressure, the safeguards systems must be capable of reducing pressure to 50% of the peak value in 24 hours. The pressure profiles developed from the M&E data are used to determine P_a to confirm that containment integrity and dose consequence limits are maintained. The analyses are discussed separately below.

LOCA Analysis for P_a

The LOCA analysis that supports the proposed change is generally consistent with the HNP license basis. The analysis uses the March 1979 model (WCAP-10325-P-A) previously reviewed and approved by the NRC for application at HNP (NUREG-1038). The postulated double-ended pump suction rupture (DEPS) and double-ended hot leg rupture (DEHL) were considered in the analysis. The DEHL has been determined to be the design basis LOCA for peak containment pressure. The re-analysis assumes complete thermal equilibrium mixing condition for the steam and emergency core cooling system (ECCS) injection water during the reflood phase for each loop receiving ECCS water. This differs from the HNP analysis of record (WCAP-10326-A, May 1983), but is consistent with the usage and application of this mass and energy (M&E) release evaluation model in subsequent analyses, for example, D. C. Cook Docket No. 50-315, June 9, 1989.

A reactor core power of 2900 MWt is assumed for the analysis. A higher initial power level results in increased core-stored energy and associated decay heat power, and higher RCS temperatures yield a higher initial fluid energy. Operation at reactor core power of 2775 MWt is thus bounded by the analysis performed at reactor core power of 2900 MWt. The RCS mass available for release is based on RCS volume that accounts for thermal expansion and uncertainty and RCS pressure that includes an uncertainty allowance. Zero percent tube plugging is modeled to maximize both the RCS fluid release and the transfer of secondary heat across the SG tubes. Additionally, this assumption lowers reactor coolant loop resistance, reducing the pressure drop upstream of the break for the pump suction breaks and increasing break flow.

Since loss of offsite power is an inherent assumption for the calculation of M&E releases, the application of single failure criteria yields two DEPS analysis cases: minimum safeguards safety

injection (SI) flow (assumes loss of one emergency diesel generator) and maximum safeguards SI flow.

The computer codes used to analyze the phases of M&E release are SATAN VI, WREFLOOD, FROTH, and EPITOME. Energy sources include the core, primary fluid and metal mass, secondary fluid/steam and metal mass, accumulator and pumped SI water, and secondary energy transfer. The decay heat model is the ANS Standard-5.1, approved by the NRC in WCAP-10325-P-A in May 1983.

The CONTEMPT-LT/26 computer code was used to determine the peak containment pressure and pressure profile that result from the M&E releases. The peak calculated pressure (P_a) is 41.8 psig (56.4 psia) and the maximum pressure at 24 hours following the LOCA is 7.9 psig (22.5 psia).

The NRC acceptance criteria presented in SRP 6.2.1.3 reflect the requirements of 10 CFR 50, Appendix A and Appendix K, paragraph I.A. In satisfaction of these requirements, the analysis addresses break size and location, calculation of each accident phase, and sources of energy. P_a, including appropriate uncertainties, is lower than the design pressure (45 psig) and the containment pressure at 24 hours following the event satisfies the 50% criterion.

MSLB Analysis for P_a

The evaluation model used for the calculation of MSLB M&E releases inside and outside containment is an approved Westinghouse methodology. The license basis methodology of WCAP-8822 (1976) has been updated for the current analysis by the use of WCAP-8822-S1-P-A and WCAP-8822-S2-P-A (1986) and WCAP-10961, Rev.1 (1985). The updated WCAPs document the use of the LOFTRAN code instead of the older MARVEL code. Consistent with the HNP license basis, this model does not assume superheated steam in the M&E from the faulted steam generator. The analysis considers a range of conditions for plant operation, safety system performance, and pipe break size. These cases bound those presented in the FSAR.

Nominal reactor core power is assumed to be 2900 MWt and, since a MSLB can occur during operating conditions ranging from full power to hot shutdown, cases at plant power levels of 102 percent, 70 percent, 30 percent and 0 percent are considered. For each power level, two break sizes are defined: A double-ended rupture (DER) downstream of the flow restrictor in one steamline, and a small split rupture that will neither generate a steamline isolation signal from the Engineered Safety Features Actuation System nor result in water entrainment in the break effluent.

Plant initial conditions are generally assumed to be the nominal values corresponding to the initial power level for that case, with appropriate uncertainties included. Key inputs are RCS pressure and RCS average temperature, pressurizer water volume, steam generator water level, steam generator pressure, and feedwater enthalpy. An RCS average temperature at the high end

of the T_{avg} range is assumed because the mass and energy releases bound those at lower temperatures. Expected values that are conservative for peak pressure are chosen for initial containment pressure (maximum) and humidity (minimum). Sufficient main feedwater flow to match or exceed the steam flow prior to reactor trip is assumed. Feedwater temperatures appropriate to the power level are selected to ensure that the mass added to the faulted SG is conservative.

Initial condition uncertainties are applied to the RCS average temperature (6°F), the steam generator mass (8 percent narrow-range span), and the power fraction (2 percent) and feedwater temperature (2°F) at full power. Uncertainty conditions are applied to those parameters that could increase the amount of mass or energy discharged into containment.

Postulated single failures include faulted loop main feedwater isolation valve (MFIV), faulted loop main steam isolation valve (MSIV), faulted loop main feedwater control valve (MFCV), auxiliary feedwater (AFW) pump runout control, and one of two trains for containment cooling. These are grouped into three cases: 1) failures of MFIV, MSIV, and AFW pump runout, 2) failures of MSIV, MFCV, and AFW pump runout, 3) failures of MSIV, one cooling train, and AFW pump runout.

Containment pressure profiles were generated using a validated version 1.0 of the CONTEMPT-LT/28 computer code. The calculated peak pressure value of 41.3 psig results from a DER at 30% power, is lower than the design pressure of 45 psig, including appropriate uncertainties, and is less than the calculated peak pressure for a postulated LOCA.

SG Water Level/RC Loop Operability

Reactor coolant loop operability in Modes 3, 4, and 5 requires that an adequate heat sink be provided by the SG secondary for removal of decay heat. A water level above the top of the SG tube bundle is needed to ensure an adequate heat sink. Since the elevation of the level instrument taps relative to the top of the tube bundle is different for the RSG, the values that ensure an appropriate minimum water level must be revised.

Technical Specification 3/4.4.1 specifies the minimum SG level for a given Mode, currently in terms of percent narrow range span (%NR). The required NR level is revised to reflect the RSG design differences. A minimum indicated level for the wide range (WR) SG level channels has been determined for Modes 4 and 5 and the WR indication will be used for verifying adequate heat sink unless the channel is inoperable. An uncertainty allowance is added to the indicated top-of-tubes value to assure a conservative %NR or %WR limit.

Technical Specification 3.4.1.2 requires two operable RC loops and at least one RC pump in operation during Mode 3 with the reactor trip system (RTS) breakers open (two RC pumps with the RTS breakers closed). Existing Special Test Exception Technical Specification 3.10.4.b requires only two operable RC loops (i.e., no operating RC pumps) during the performance of

hot rod drop time measurements. This exception and associated surveillance requirement are not needed at HNP because rod drop testing is performed only with all three RC pumps operating to simulate a reactor trip under actual conditions.

Borated Water Source

Based on analysis results, the minimum contained volume during shutdown specified for the boric acid tank in Technical Specification 3.1.2.5 is increased from 6650 to 7150 gallons (associated level changes from 21 to 23%). This change accounts for the increased RCS volume due to the RSGs, a bounding "equilibrium fuel cycle", and other analysis assumptions.

Group II **RTS/ESFAS and Core Limits**

TS #	Title
Fig. 2.1-1	Core Safety Limit Line Plot
B 2.1.1	
3.2.5.a	DNB Parameters
B 3/4.2.5	
6.9.1.6	Core Operating Limits Report
Table 2.2-1	RTS Instrument Trip Setpoints
Table 3.3-4	ESFAS Instrument Trip Setpoints

BACKGROUND

HNP was designed and initially operated at a nominal average reactor coolant temperature (T_{avg}) of 588.8°F at Rated Thermal Power (RTP). In an effort to extend the service life of the Westinghouse Model D4 steam generators, T_{avg} was reduced by Technical Specification Amendment No. 46 from 588.8°F to 580.8°F. Related changes to reflect the lower T_{avg} were made to the maximum T_{avg} value in Technical Specification 3.2.5.a and to Technical Specification Table 2.2-1 Notes 1 and 3 associated with the trip setpoints for overtemperature ΔT (OTΔT) and overpower ΔT (OPΔT). The amendment also included changes to the reactor core safety limits, Technical Specification Figure 2.1-1, related to the transition from reactor fuel supplied by Westinghouse (LOPAR and Vantage 5) to reactor fuel supplied by Siemens Power Corporation(HTP).

Core Safety Limits and DNB Parameters

The design departure-from-nucleate-boiling ratio (DNBR) is protected by reactor core safety limits. These plots (Technical Specification Figure 2.1-1) provide limits for the combination of thermal power, RCS pressure and T_{avg} to prevent overheating of the fuel and possible cladding perforation caused by exceeding the DNBR limit. The maximum T_{avg} value in Technical Specification 3.2.5 (DNB Parameters) is an analytical limit to ensure that the minimum DNBR is equal to or greater than its design value by maintaining T_{avg} within the normal steady state envelope assumed in transient and accident analysis. Technical Specification 6.9.1.6 provides a list of NRC-approved references for the analytical methods used to determine the core operating limits.

RTS/ESFAS Tables

The instrumentation and interlocks of the reactor trip and engineered safety features actuation system (RTS/ESFAS) ensure that the associated reactor trip and/or safeguards actions are initiated when the monitored parameter(s) reaches the trip setpoint(s). The trip setpoints listed in Technical Specification Tables 2.2-1 and 3.3-4 are nominal values that: ensure the core and reactor coolant system do not exceed their safety limits during normal operation and design basis accidents; and initiate safeguards actions for mitigating accident consequences.

BASIS FOR PROPOSED CHANGE

The reactor core safety limits are protected by the overtemperature delta temperature (OTΔT) and overpower delta temperature (OPΔT) trip setpoints, a part of the RTS. The OTΔT trip, Table 2.2-1 Functional Unit 7, prevents DNB for all combinations of pressure, power, coolant temperature and axial power distribution. The OPΔT trip, Table 2.2-1 Functional Unit 8, provides assurance of fuel integrity (i.e., no melting of fuel pellets and less than 1% cladding strain) under all overpower conditions. The setpoint ensures that the allowable heat generation rate (kW/ft) is not exceeded and limits the required range for OTΔT trip.

The allowable values account for measurement and calibration accuracies and the instrument drift expected to occur between surveillance tests. Operation with setpoints that are less conservative than the nominal setpoint value but within the allowable value is acceptable since the value used in the safety analysis is not exceeded. The TA, Z and S terms are used in determining channel operability if the "as-measured" trip setpoint is found to exceed the allowable value. A determination of operability can only be made if the safety analysis value remains bounding.

The calculated values for RTS/ESFAS trip setpoints, allowable values, and associated terms are based on Westinghouse setpoint methodology approved in the original HNP license basis and are consistent with the requirements of ISA Standard 67.04.

The RTS Functional Units in Table 2.2-1 associated with SG level are (13) SG water level low-low and (14) SG water level-low coincident with steam/feedwater flow mismatch. The low-low SG water level trip protects the reactor from loss of heat sink and is credited in accident analyses for reactor trip. The low SG water level trip coincident with steam/feedwater flow mismatch is not credited in the transient and accident analysis but enhances the overall reliability of the reactor trip system.

The ESFAS Functional Units in Table 3.3-4 associated with SG level are (5.b) SG water level high-high (for turbine trip and feedwater isolation) and (6.c) SG water level low-low (for auxiliary feedwater). If the SG level ESFAS setpoint is exceeded and other necessary signals received to satisfy the required logic combination, the related safeguards action is initiated.

PROPOSED CHANGES

Technical Specification Figure 2.1-1 and Bases are revised to reflect reactor core safety limit plots developed by Siemens. Core loads following SG replacement will consist entirely of Siemens fuel.

Technical Specification 3.2.5.a and Bases are revised by changing the limit on T_{avg} from ≤ 586.1°F to ≤ 594.8°F to reflect the 8°F increase in nominal T_{avg} and the current analysis value for RCS average temperature uncertainty.

BASIS FOR PROPOSED CHANGE

Technical Specification 6.9.1.6 is revised to reference the appropriate analytical methods used to determine the core operating limits in support of SG replacement and the use of all-Siemens fuel.

Technical Specification Table 2.2-1 and 3.3-4 setpoints and related values are revised to reflect RSG design differences, margin improvement analysis for OTΔT and OPΔT setpoints, increases to safety analysis limits, updated channel uncertainty calculations, and supporting analytical results.

BASIS

Core Safety Limits and DNB Parameters

The existing reactor core safety limits are based on transition core loads consisting of both Westinghouse and Siemens fuel. The revised limits are for cores consisting of only Siemens fuel using the approved Siemens HTP correlation. These limits are appropriate for operation at either the current licensed reactor core power level (2775 MWt) or at uprated reactor core power level (2900 MWt). The Technical Specification Bases section related to core safety limits and the Technical Specification administrative section for the Core Operating Limits Report are revised to reflect a Siemens-only core and methodologies.

Replacement of the HNP SGs eliminates the need to operate at the reduced T_{avg} since the RSGs are designed to operate at a T_{avg} of 588.8°F without concern for tube degradation experienced with the OSGs. The original T_{avg} value of 588.8°F, with an appropriate uncertainty value, is used as an input to the transient and accident analyses performed in support of the SGR. The results demonstrate that the proposed T_{avg} limit of 594.8 is adequate to maintain a minimum DNB Ratio (DNBR) at or above the design DNBR value during normal operation and analyzed transients.

RTS/ESFAS Tables

The proposed revision to RTS/ESFAS table values can be grouped into the following categories to describe the bases for the changes.

Physical differences between the RSG and the OSG, such as NR instrument tap elevations, primary/secondary volumes and heat transfer characteristics, require changes to certain trip setpoints and allowable values. The setpoints and associated values for SG water level trip functions are affected by these differences. The determination of channel uncertainties for the revised setpoints is consistent with the replacement design and Westinghouse setpoint methodology approved for HNP.

OTΔT/OPΔT setpoint terms and time constants are changed as a result of applying Westinghouse margin improvement methodology previously approved for Farley Nuclear Plant. The analysis performed for HNP eliminates Lead-Lag compensation (setting τ_1 and τ_2 to zero seconds) and provides a filter for spurious temperature fluctuations by increasing Lag compensation (setting τ_3 to 4 sec.).

Using a conservative simulation of ΔT fluctuation that bounds typical plant data, adequate margin to trip has been demonstrated. Transient and accident analysis results support adequate DNBR margin for those events mitigated by the OTΔT trip and margin to fuel melt and cladding strain for those overpower events mitigated by the OPΔT trip. The safety analyses and results support the proposed Technical Specification changes and confirm that the DNB design criterion is met.

Changes to total allowance (TA) terms reflect increases to safety analysis limits (SALs) used for the analyses. Proposed TA values for OTΔT, OPΔT, the nuclear instrumentation channel for high negative rate, and high containment pressure are examples of this type of change.

Changes to RTS/ESFAS table values for Z , S, and allowable value (AV) are based on instrument channel uncertainty calculations. These updated uncertainty calculations were prepared in accordance with the guidance of ISA S67.04-1994 and are consistent with the existing HNP licensing basis.

The results of the transient and accident analyses and evaluations demonstrate that applicable acceptance criteria continue to be met. Setpoint uncertainty calculations confirm acceptability of the proposed trip setpoint values. The changes to other RTS/ESFAS table values reflect the results of calculations based on RSG process measurement uncertainties and HNP-specific uncertainties, calibration practices and equipment, installed hardware and procedures.

Group III Dose and Specific Activity

TS #	Title
3.4.8 Fig. 3.4-1 Table 4.4-4 B 3/4.4.8 1.11	Specific Activity Definitions – Dose Equivalent I-131

BACKGROUND

The Technical Specification 3/4.4.8 limits on RCS specific activity impact assumptions used in accident dose calculations. In general, RCS activity affects the determination of dose consequences for events that release RCS inventory to containment or directly to the environment, such as steam generator tube rupture (SGTR).

For the SGTR event, the reactor coolant specific activity limits ensure that the resulting 2-hour dose at the site boundary will not exceed a small fraction of 10 CFR 100 dose guidelines assuming an accident initiated spike, and will also not exceed 10 CFR 100 guidelines for the pre-accident iodine spike cases. The SGTR analyses demonstrate that the limiting specific activity values, in conjunction with an assumed 1 gpm steady-state primary-to-secondary SG leakage rate, assure that potential radiological consequences from the design basis event are acceptable.

The existing Technical Specification limit for Dose Equivalent Iodine 131 (DEI -131) of 1.0 microcurie per gram (μCi/gm) represents the threshold for changing from normal frequency surveillance to increased frequency sampling. Should the RCS DEI-131 inventory remain above 1.0 μCi/gm for 48 hours, Technical Specification require the plant to be shut down. In addition, the limit for a single analysis determination as a function of rated thermal power (RTP) is specified in Figure 3.4-1. As shown in the figure, the limit is a constant 60 μCi per gram from 100% to 80% RTP, then increases linearly above that value with decreasing RTP.

The thyroid dose conversion factors used for the calculation of dose-equivalent I-131 are those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

PROPOSED CHANGES

Technical Specification 3/4.4.8, Table 4.4-4 and associated Technical Specification Bases are revised to specify 0.35 μCi per gram for the 48-hour interval analysis limit and 60 μCi per gram at all power levels for a single sample analysis limit. Figure 3.4-1 is deleted.

Technical Specification 1.11 definition of dose equivalent I-131 is modified to reflect, as the basis for thyroid dose conversion factors, the International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, Volume 3 No. 1-4, 1979 (or equivalently, Federal Guidance Report (FGR) No. 11, Limiting Values of

Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, EPA 520/1-88-020, dated: September 1988).

BASIS

The Project dose analyses update the methods used in the analysis of record (AOR) by incorporating approved modeling techniques (e.g., eliminating the penalty for tube uncovery during SGTR) and conservative assumptions for operating conditions (e.g., uprated power level). Based on the current NEI 99-03 industry issue regarding control room (CR) habitability calculations that use very low unfiltered in-leakage values, Project dose evaluations use a value of in-leakage significantly higher than the AOR. HNP CR ventilation calculations were reviewed and a value conservative to all previous in-leakage assumptions is utilized. The current AOR in-leakage assumption is 3 cubic feet per minute (cfm), compared to the Project limiting analysis value of 45 cfm.

In general, the analyses are based on the current Technical Specification limit of 1.0 μCi/gm and are therefore conservative with respect to the proposed reduction. The only analysis that does not meet guidelines using the existing Technical Specification limit is the SGTR event. The SGTR analysis for the Project determined that a reduced initial inventory (0.35 μCi/gm) is necessary to meet, with margin, the Standard Review Plan and Regulatory Guide acceptance criteria. Details of the SGTR analysis are discussed below relative to the proposed Technical Specification change.

In the AOR, the thyroid dose conversion factors are from Regulatory Guide (RG) 1.109 (Table III of TID 14844). In order to be consistent with current analysis techniques and NRC expectations, thyroid dose conversion factors from ICRP-30 (or the essentially equivalent FGR 11) is used for the SGR/PUR analyses.

Dose Analysis Results

Although some of the analyzed events lead to dose consequences that exceed the current HNP analyses of record, all meet the appropriate SRP and Regulatory Guide acceptance criteria. Analyses inputs, assumptions, methods, and results are discussed in deta il in Section 2.22 of Enclosure 7. A summary of the results is provided in Tables 2 through 4. Values are in units of radiation equivalent-man (rem) and are specified for the exclusion area boundary (EAB) except for Table 4 main control room doses.

TABLE 2
ANALYSES WITH OFFSITE DOSE INCREASES

Accident	Acceptance Criteria[2]	FSAR Value	SGR/Uprate Value	Comments
15.4.3.2 Withdrawal of Single Full Length RCCA (III)	Thyroid < 30 rem	N/A[1]	6.18	For the dose analysis the assumed fuel failure is 4% clad/ 2% FCLM[3]
	Whole Body < 2.5 rem	N/A[1]	1.52	
15.4.7 Inadvertent Loading of Fuel Assembly (III)	Thyroid < 30 rem	N/A[1]	6.18	For the dose analysis the assumed fuel failure is 4% clad/ 2% FCLM
	Whole Body < 2.5 rem	N/A[1]	1.52	
15.4.8 Spectrum of RCCA Ejection Accidents (IV)	Whole Body < 6 rem	~0.12	~1.6	No fuel failures are calculated, but 4% clad and 2% FCLM are assumed
15.6.2 Dose Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment (II)	Whole Body < 2.5 rem	0.044	0.05	The event is a function only of the RCS noble gases and DEI-131.
15.7.1 Radioactive Waste Gas System Leak or Failure	Whole Body 0.5 rem	0.28	0.29	
15.1.5 Main Steam Line Break - Stuck RCCA – (No Iodine Spike)	Thyroid < 300 rem	46	84.5	Thyroid and Whole Body doses increased
	Whole Body < 25 rem	0.3	0.703	
15.1.5 Main Steam Line Break – Accident Initiated Iodine Spike	Thyroid < 30 rem	2.6	5.38	Note 4
15.6.3 SGTR – Pre-accident Iodine Spike	Whole Body < 25 rem	0.22	0.35	Increase due to inclusion of iodine contribution
15.6.3 SGTR – Accident Initiated Iodine Spike	Whole Body < 2.5 rem	0.22	0.26	

[1] The current HNP license basis assumes no fuel failure for these events.
[2] 10CFR100 or Standard Review Plan value
[3] FCLM – Fuel Centerline Melt
[4] The FSAR lists 48 rem (exceeding the SRP limit of 30 rem). A footnote indicates: "This dose results from inclusion of a 1% fuel failure in the radiological consequences of a main steam line break analysis. A less conservative analysis has been performed in which only the accident generated iodine spike has been considered. The resulting dose is 2.6 rem."

TABLE 3
MAIN CONTROL ROOM DOSE INCREASES

Event	Acceptance Criteria	FSAR Value	SGR/Uprate Value	Comments
15.6.5 LOCA (III & IV)	Thyroid < 30 Rem	7.1	28	SGR/Uprate MCR In-leakage assumed to be 80 CFM versus 3 CFM for FSAR
	Whole Body < 5 Rem	0.6	0.55	
	Skin < 30 Rem	14	11.3	
15.1.5 MSLB	Thyroid < 30 Rem	Not in FSAR	27	Limiting SGR/Uprate case (stuck RCCA and no pre-accident Iodine spike) is presented. No MSLB CR dose reported in the FSAR.
	Whole Body < 5 Rem		0.720	
	Skin < 30 Rem		20.2	
15.6.3 SGTR	Thyroid < 30 Rem	Not in FSAR	10.93	Pre-Accident Spike is the limiting case. Main CR In-leakage assumed to be 45 CFM vs. 3 CFM for AOR.

SGTR Analysis

The RSGs and OSGs have design differences that potentially affect SGTR analyses, such as normal water level relative to the remaining free volume, SG tube size and heat transfer area. SGTR accident analyses were performed to reflect the RSG differences and other changes. The analyses are based on methodology documented in WCAP-10698-P-A and approved by the NRC (SERs dated 12/17/85 and 3/30/87). The methodologies are developed based on the use of the LOFTTR2 computer program and have been applied in previous HNP SGTR analyses. Any variations in the methodology reflect improvements in selecting and evaluating inputs based on the latest accepted methods.

The analyses are performed in two phases that require different limiting assumptions: determination of margin to SG overfill; and evaluation of radiological consequences, including radiation dose at the site boundary, radiation dose at the low population zone, and radiation dose to control room personnel. The design basis SGTR assumes a double-ended tube break located at the tube sheet on the outlet (cold leg) side of the SG, loss of offsite power and a fully withdrawn control rod assembly at reactor trip. A range of main feedwater temperatures (375°F to 440°F) was modeled to support a large operating window.

SGTR Margin to Overfill

Margin to overfill is demonstrated in the first phase of the analysis and precludes the impacts of overfill from the dose consequences. The analysis assumes the following plant conditions: T_{avg} of 588.8°F, main feedwater temperature of 375°F, initial SG secondary mass greater than nominal (+10%), and a tube plugging maximum value of 10%. The limiting single failure was determined to be a power operated relief valve (PORV) failure (closed) on an intact SG. Differences in conditions or assumptions from those in the approach previously approved for HNP provide additional conservatism, such as: turbine runback is assumed as before, increasing secondary mass, but without taking credit for the associated delay in reactor trip; and immediate maximum auxiliary feedwater (AFW) flowrate, without taking credit for startup delays.

Changes were made to the license basis assumptions for the sequence of operator actions to ensure adequate margin to overfill in the Project analysis. In accordance with the guidance of NRC Information Notice 97-78, these changes are discussed below.

The operator actions required to stabilize plant response and terminate primary-to-secondary break flow in the event of an SGTR are identified in revised HNP emergency operating procedures (EOPs). In conjunction with updating the EOPs to Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERG) Rev. 1.C, appropriate operator actions and assumed times have been developed as a part of the margin to overfill analysis. These operator actions, including isolating the ruptured SG, initiating RCS cooldown and depressurization, and terminating safety injection have been validated by HNP operating crew simulator runs. A comparison of key actions and corresponding times assumed for the analysis of record (AOR) vs. the Project analysis are shown in the table below. The results demonstrate that SG overfill will not occur.

TABLE 4

OPERATOR ACTION TIMES FOR SGTR ANALYSIS

Action	SGR/Uprate Analysis	Analysis of Record
Isolate auxiliary feedwater flow to ruptured SG	Maximum of 10 min. or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG	Maximum of 13 min. 35 sec. Or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG
Isolate steam flow from ruptured SG	Maximum of 12 min. or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG	Maximum of 13 min. 35 sec. Or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG
Operator action time to initiate cooldown	5 min. from time of steamline isolation*	8 min. 1 sec. From time of steamline isolation
Cooldown	Calculated by LOFTTR2	Calculated by LOFTTR2
Operator action time to initiate depressurization	4 minutes from end of cooldown	2 min. 16 sec. From end of cooldown
Depressurization	Calculated by LOFTTR2	Calculated by LOFTTR2
Operator action time to terminate SI following depressurization	Maximum of 3 minutes from end of depressurization or time to satisfy termination criteria	Maximum of 3 minutes from end of depressurization or time to satisfy termination criteria
Pressure equalization	Calculated by LOFTTR2	Calculated by LOFTTR2

*The steamline isolation step is retained to model the time when the operators reach the EOP step requiring isolation of all flow in and out of the ruptured steam generator, although isolation may occur automatically due to safeguards response to reactor trip and loss of offsite power.

SGTR Dose

The following plant conditions were assumed for determining offsite dose mass releases - T_{avg} (588.8°F), main feedwater temperature of 375°F, initial SG secondary mass less than nominal (-10%), and a tube plugging minimum value of 0%. The limiting single failure is a failed open PORV for the ruptured SG. Other conditions and assumptions used for the margin to overfill analysis are also conservative in determining dose. Differences assumed for the dose analysis include no turbine runback (less secondary mass and early reactor trip); minimum AFW flow rate; and higher flashing fraction by assuming all flow through the hot leg side of the break.

The operator actions and response times assumed in the dose analysis are similar to those discussed for margin to overfill except for the need to isolate the failed open PORV on the ruptured SG. This is modeled in the analysis by locally closing the associated block valve at a time of 20 minutes after the PORV failure.

For the dose calculation, source term values for the SGTR pre-accident and accident-initiated iodine spikes are set at their respective proposed Technical Specification limits. Separate

evaluations of dose against the SRP dose limits were performed for the two separate Technical Specification basis conditions. The Nuclear Safety Advisory Letter NSAL-00-004, March 2000, provides recent Westinghouse guidance for evaluating the impact of these Technical Specification limits on the Westinghouse dose analysis methods. The NSAL discusses additional leakage and cleanup path assumptions that affect the "appearance rate" of components of the DEI-131 values assumed in the analysis.

These assumptions provide additional "removal terms" for the production-removal equilibrium assumed to exist prior to the accident. The significant increases in the amount of DEI-131 removal result in a similar increase in the "appearance rate". The appearance rate is multiplied by a factor of 500 (for the accident initiated spike case) in accordance with SRP analysis methodology. This has a major influence on the dose consequence results associated the current Technical Specification limit of 1.0μCi/gm. A longer spike duration is modeled in response to NRC comments on recent analyses performed for other plants. A "sensitivity study" was performed as part of the event analysis that showed that analysis inputs associated with the RSG or uprated power option had little net impact on the results.

Initial RCS concentration of noble gases is based on one percent failed fuel and is equal to the Technical Specification limit of 100/$\bar{E}$. The dose conversion factors used to calculate initial iodine concentrations in the existing analysis of record (AOR) are based on Regulatory Guide (RG) 1.109. This basis has been updated in the current analysis and is consistent with International Commission on Radiological Protection (ICRP-30) factors. The partition factors used in the iodine transport model are consistent with NUREG-0800 SRP 15.6.3.

Other assumptions used in the proposed new analysis that are different from the AOR include neglecting the effect of iodine scrubbing of the steam bubbles rising from the rupture site; not crediting radioactive decay or ground deposition; and, as a result of NRC approval of the Westinghouse Owners Group (WOG) resolution, not including a penalty for tube uncovery.

The Project dose analyses do not consider the effect of reduced power on the source terms or release paths associated with SGTR. Since the analyses do not specifically evaluate and justify an increase in allowable pre-event spiking at reduced power (existing Technical Specification Figure 3.4-1), this power dependence is eliminated and the single, constant value of 60μCi/gm for all power levels is proposed.

Conclusion

All dose consequences at the exclusion area boundary, low population zone, and control room determined by the proposed Specific Activity Technical Specification change and Project analyses are within SRP guidelines and 10CFR100 limits.

Group IV Plant Systems & Other

TS #	Title
Table 3.7-1	Max Power Range Neutron Flux High Setpoint w/ inoperable MSSVs
B 3.7.1.3	Condensate Storage Tank
3/4.4.5 B 3/4.4.5	SG Surveillance Requirements
B 3/4.7.2	SG Pressure/Temperature Limitation
B 3/4.7.1.2	Auxiliary Feedwater System

BACKGROUND

Power Range Neutron Flux Monitor Setpoints

Operability of all the main steam line Code safety valves (MSSVs) ensures that the secondary system pressure will be limited to less than 110% of its design pressure during the most severe anticipated operational transient. The maximum relieving capacity is sufficient to accommodate a turbine trip from 100% rated thermal power (RTP) coincident with loss of condenser heat sink. With less than full MSSV capacity available (ie, one or more valves inoperable), operation may proceed only at reduced power levels. Technical Specification 3.7.1.1 specifies reduced reactor trip setpoint values for the power range neutron flux channels to satisfy the safety limit for main steam pressure for up to 3 inoperable safety valves on each operating SG.

Condensate Storage Tank

The condensate storage tank (CST) is the primary source of supply for the auxiliary feedwater system. The minimum stored volume allowed by Technical Specification Bases 3.7.1.3 of 270,000 gallons is sufficient to provide auxiliary feedwater for hot standby, depressurization and cooldown needs to reach conditions suitable for initiating operation of the RHR system. The associated Bases statement indicates that hot standby conditions can be maintained for 12 hours with steam discharge to the atmosphere concurrent with total loss of power.

SG Surveillance

Technical Specification 3.4.5 provides tube inspection criteria and surveillance requirements for the OSGs. The criteria and related plugging and repair requirements provide reasonable assurance that the structural and leakage integrity of the tubes is acceptable for continued service and there is an adequate margin of safety to withstand the loads imposed during normal operation and postulated accidents.

The current in-service inspection program is needed to detect evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or in-service conditions that could lead to tube degradation. In-service inspection also provides a means of characterizing the nature and cause of tube degradation so that corrective measures can be taken. For the SGR outage, the in-service inspection is not required because the SG is new and pre-service testing includes hydrostatic pressure testing and eddy current baseline examinations.

BASIS FOR PROPOSED CHANGE

The current repair method of tube sleeving provides a process to re-establish the primary system pressure boundary for tubes that meet repair criteria. Installation of the sleeve provides a leak tight boundary that spans the defective area and restores the structural integrity of the tubing to satisfy RG 1.121 requirements. The repair criteria in the existing Technical Specification are based on analyses performed to requirements of regulatory guide (RG) 1.121. Prior to industry development of the associated repair methods, SG tube plugging criteria were based solely on a tube integrity minimum wall thickness.

Reporting requirements associated with application of F* repair criteria provide assurance that the NRC is aware of any abnormalities detected during an inspection and requires notification as to the extent of application of the criteria.

SG Pressure/Temperature Limits

The Bases for Technical Specification 3/4.7.2 states that the limitations on SG pressure and temperature (200 psig and 70°F, respectively) are imposed to ensure that the maximum allowable fracture toughness limits are not exceeded and are based on a SG RT_{NDT} of 60°F.

Auxiliary Feedwater Pump Performance

The Bases for Technical Specification 3/4.7.1.2 indicates that the auxiliary feedwater (AFW) pump performance requirements that satisfy accident analysis assumptions for AFW flow are based on a 4% degradation of the vendor certified performance curves.

PROPOSED CHANGES

Technical Specification Table 3.7-1 is revised to reflect new maximum allowable setpoint values for the power range neutron flux channels when main steam line safety valves are inoperable.

Technical Specification B3.7.1.3 is revised to indicate that the condensate storage tank minimum water volume is sufficient to maintain the RCS at hot standby for 6 hours with steam discharge to the atmosphere concurrent with loss of offsite power.

Technical Specification 3/4.4.5 is revised to reflect replacement of the OSGs with RSGs. The text that addresses F* criteria, SG tube repair, and sleeve/preheater inspections is related to types of degradation associated with the OSGs and is not applicable to the RSGs.

Technical Specification Bases 3/4.7.2 for SG Pressure/Temperature limitations is revised to clarify the basis for the limitations and to indicate the RT_{NDT} value for the replacement SGs.

Technical Specification 3/4.7.1.2 is revised to indicate that the AFW pump performance requirements that satisfy accident analysis assumptions for AFW flow are based on a maximum allowable degradation of pump performance curves.

BASIS

Power Range Neutron Flux Monitor Setpoints

Technical Specification Table 3.7-1 specifies maximum allowable values for the power range neutron flux high setpoint if one or more safety valves are inoperable on an operating SG. The reduced setpoint values ensure that the total relieving capacity of the remaining operable safety valves will limit the secondary system pressure to 110% of design during postulated transients. The maximum values are specified as a percentage of RTP and are determined based on the methodology described in Technical Specification Bases. The algorithm used in calculating the reduced setpoints contains the plant nominal NSSS power rating in the denominator. An uprate reactor core power level of 2900 MWt was used in the calculation, resulting in a reduction in the setpoint values. The lower values are conservative for operation at the current reactor core power level of 2775 MWt and support the option of operating at uprated power.

Condensate Storage Tank

The requirement for the condensate storage tank is that the minimum seismic category I volume is sufficient to provide auxiliary feedwater for four hours at hot standby followed by the longest time expected for cooldown to RHR entry conditions. The most limiting method for RCS depressurization and a conservative cooldown rate (with loss of offsite power) is assumed. The existing Technical Specification minimum CST volume is 270,000 gallons, providing a usable volume of 235,000 gallons.

The Project analysis determined that the existing Technical Specification required minimum CST volume has adequate margin to support a period of 6 hours at hot standby, which exceeds the minimum SRP requirement. Accordingly, Technical Specification 3.7.1.3 is unaffected and the Bases statement will be revised to reflect a capability for a 6-hour period at hot standby with steam discharge to the atmosphere concurrent with loss of offsite power.

SG Surveillance

The program for periodic in-service inspection of the SGs monitors the integrity of the SG tubing to provide reasonable assurance that proper and timely corrective action can be taken in the event of tube degradation. The existing Technical Specification sleeving methodologies, F* criteria, and sleeve/preheater region inspection requirements were developed specifically to address the types of tube degradation found in the OSGs and other early designs and are not applicable to the RSGs. RSG materials and design features for tubing, tube supports and feedwater are different from the OSGs. In light of the design differences, text associated with F* repair criteria, sleeving, and preheater region inspections should be removed. The proposed Technical Specification acceptance limits for SG tubes are based on through-wall criteria that require tube plugging when imperfections exceed the plugging limit of 40% of the nominal tube wall thickness.

BASIS FOR PROPOSED CHANGE

The Technical Specification plugging limit, in conjunction with surveillance and maintenance programs, provides reasonable assurance that the SG tubes accepted for continued service will retain adequate structural and leakage integrity during normal, transient, and postulated accident conditions. This is consistent with general design criteria (GDC) 14, 15, 30, 31, and 32 of 10 CFR 50, appendix A. Compatibility with these GDCs supports the application of Technical Specification acceptance limits based on through-wall criteria.

The criterion for tube plugging is based on a minimum wall thickness due to wastage as determined by ASME Section XI. This change is conservative in that the possibility of operation with F* indications is eliminated.

Therefore, this proposed change will provide reasonable assurance that the HNP SGs maintain adequate structural and leakage integrity consistent with applicable GDCs and 10 CFR 50, appendix A.

SG Pressure/Temperature Limits

The RT_{NDT} for the RSG is limited to a maximum of 10°F. Therefore, the existing pressure and temperature limits, based on an RT_{NDT} of 60°F, are conservative.

Auxiliary Feedwater Pump Performance

The methodology for determining minimum allowable AFW pump performance is changed from a percentage of the vendor pump curve to a maximum allowable degradation. Since, the proposed degradation methodology maintains the existing surveillance minimum requirements, minimum allowable pump performance is unchanged and continues to satisfy analysis assumptions.

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
TECHNICAL SPECIFICATION CHANGE REQUEST
STEAM GENERATOR REPLACEMENT

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (10 CFR 50.92)

(17 pages total)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (10 CFR 50.92)

The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Carolina Power & Light Company has reviewed this proposed license amendment request and determined that its adoption would not involve a significant hazards determination.

BACKGROUND

Carolina Power & Light (CP&L) has completed a comprehensive engineering review program in support of Harris Nuclear Plant (HNP) plans to replace the existing steam generators, Westinghouse Model D4 (OSG), with Westinghouse Model Delta 75 (RSG). The program considers RSG-OSG differences and an increase to the current nominal average reactor coolant temperature (T_{avg}) to the original T_{avg} of 588.8°F. A single set of analyses that encompasses steam generator replacement (SGR), a range of operating conditions, and the power uprate option was performed to provide conservative results that bound operation at the current power level following replacement. Although the analyses results support operation at both existing licensed reactor core power (2775 MWt) and uprated reactor core power (2900 MWt), this submittal does not include a request for power uprate.

The engineering review program, hereinafter referred to as the Project, includes evaluation of impacts to NSSS and BOP systems, identification of required modifications, and re-analysis or evaluation of design transients and postulated accidents. The results of the review are described in the NSSS and BOP License Reports and the results demonstrate applicable acceptance criteria continue to be met.

Major NSSS and BOP systems and sub-systems (e.g., safety injection, residual heat removal; auxiliary feedwater, component cooling water) have been reviewed and acceptable performance verified for normal operational functions and, as applicable, for safety related functions. All reactor trip and engineered safety features actuation setpoints have been assessed and the proposed setpoint values assure adequate protection is afforded for design basis events. Control systems, including pressurizer level and SG level, have been evaluated and found acceptable for operation with the RSGs and planned modifications. Proposed changes are reflected in the re-analysis or evaluation of FSAR Chapter 15 events and transients and acceptance criteria are satisfied.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (10 CFR 50.92)

PROPOSED CHANGES

The changes are listed in the order in which they are described in the groupings below.

Harris Nuclear Plant (HNP) proposes to modify Technical Specification 5.4.2 following SG replacement to reflect the RCS volume increase due to the larger tube bundle of the RSGs. The proposed value of "approximately 10,300 cubic feet" is specified at a nominal T_{avg} of 588.8°F. The plant will be operated at the original HNP design basis value for T_{avg} of 588.8°F.

Technical Specification 6.8.4.k is modified to indicate peak calculated pressures for the design basis MSLB and LOCA of 41.3 psig and 41.8 psig (respectively). The LOCA value will be indicated for peak pressure (P_a) in Technical Specification Bases 3/4.6.1.4 and 3/4.6.1.6 to reflect the results of accident analyses performed for the Project.

Technical Specification 3/4.4.1 and associated Bases are modified to provide a revised SG minimum level limit of 30% narrow range (NR) for Modes 3, 4, and 5 and to specify that a SG minimum level limit of 74% wide range (WR) shall be used for Modes 4 and 5, unless inoperable, to support limiting conditions for operation and surveillance requirements for the RSGs.

Technical Specification 3/4.10.4 is modified to delete an unnecessary test exception for the performance of hot rod drop time measurements.

Technical Specification 3.1.2.5 and Bases are revised to specify a minimum contained borated water volume for the boric acid tank of 7,150 gallons and associated indicated level of 23%.

Technical Specification Figure 2.1-1 and Bases are revised to reflect curves developed by Siemens using the approved high thermal performance (HTP) correlation. Core loads following SG replacement will consist entirely of Siemens fuel.

Technical Specification 3.2.5.a and Bases are revised by changing the limit on T_{avg} from ≤ 586.1°F to ≤ 594.8°F to reflect the 8°F increase in nominal T_{avg} and the current analysis value for RCS average temperature uncertainty.

Technical Specification 6.9.1.6 is revised to reference the appropriate analytical methods used to determine the core operating limits in support of SG replacement and the use of all-Siemens fuel.

Technical Specification Table 2.2-1 and 3.3-4 setpoints and related values are revised to reflect RSG design differences, margin improvement analysis for over-power and over-temperature delta temperature setpoints, increases to safety analysis limits, updated channel uncertainty calculations, and supporting analytical results.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (10 CFR 50.92)

Technical Specification 3/4.4.8, Table 4.4-4 and associated Technical Specification Bases are revised to specify 0.35 microcuries per gram (μCi/gm) dose equivalent iodine 131 (DEI-131) for the 48-hour interval analysis limit and 60 μCi/gm at all power levels for a single analysis limit. Figure 3.4-1 is deleted.

Technical Specification 1.11 definition of dose equivalent I-131 is modified to reflect, as the basis for thyroid dose conversion factors, the International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, Volume 3 No. 1-4, 1979 (or equivalently, Federal Guidance Report (FGR) No. 11, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, EPA 520/1-88-020, dated: September 1988).

Technical Specification Table 3.7-1 is revised to reflect new maximum allowable setpoint values for the power range neutron flux channels when one or more main steam line safety valves are inoperable.

Technical Specification B3.7.1.3 is revised to indicate that the condensate storage tank minimum water volume is sufficient to maintain the RCS at hot standby for 6 hours with steam discharge to the atmosphere concurrent with loss of offsite power.

Technical Specification 3/4.4.5 is revised to reflect replacement of the existing Westinghouse Model D4 SGs with Model Delta 75 SGs. The deleted text addresses F* criteria, SG tube repair (i.e., sleeving), and specific inspections of sleeves and the OSG preheater region.

Technical Specification Bases 3/4.7.2 for SG Pressure/Temperature limitations is revised to clarify the basis for the limitations and to indicate the RT_{NDT} value for the replacement SGs.

Technical Specification Bases 3/4.7.1.2 is revised to provide a different methodology for determining maximum allowable AFW pump performance degradation.

Associated editorial changes to revise the table of contents for consistency are indicated in the Technical Specification mark-ups.

The evaluation of these changes with respect to 10 CFR 50.92 is provided in the following four groups:

I. Analysis Limits for RCS and Containment
II. Core Safety Limits and Reactor Trip System/Engineered Safety Feature Actuation System Trip Setpoints and Associated Values
III. Specific Activity and Dose
IV. Plant Systems and Other

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (10 CFR 50.92)

Group I Analysis Limits for RCS & Containment

TS #	Title	Description
5.4.2	RCS Volume	Change RCS volume to reflect RSG design differences and T_{avg} = 588.8°F
6.8.4.k B 3/4.6.1.4 B 3/4.6.1.6	Containment Leak Test Program Containment Systems	P_a = 41.8 psig
4.4.1.2.2 4.4.1.3.2 3.4.1.4.1.b 4.4.1.4.1.1 B 3/4.4.1	RCS – Hot Standby RCS – Hot Shutdown RCS – Cold Shutdown RC Loops and Coolant Circulation	Minimum SG level 30% NR Minimum SG level 30% NR or 74% WR Describes requirement for adequate heat sink
3.4.1.2 3.10.4 4.10.4.3	RCS-Hot Standby Special Test Exceptions – RC Loops	Unnecessary test exception deleted
3.1.2.5 B 3/4.1.2	Borated Water Source - Shutdown	Minimum water volume to ensure shutdown margin

RCS Volume (5.4.2)

The value in the Design Features section for total water and steam RCS volume is changed to reflect the increase due to installation of the RSGs and is calculated at the proposed T_{avg} of 588.8°F. Comprehensive analyses and evaluations demonstrate that plant systems, components, and structures continue to satisfy design requirements under normal and accident conditions and applicable acceptance criteria continue to be met.

Containment Systems and Containment Leakage Rate Testing Program (B 3/4.6.1.4, B 3/4.6.1.6, 6.8.4.k)

The Containment Systems Bases pages and the Containment Leakage Rate Testing Program changes for internal pressure response to LOCA and MSLB events reflect results of containment analyses performed for operating conditions that include uprated power with the RSGs. The calculated LOCA peak pressure (P_a) of 41.8 psig exceeds the MSLB peak pressure and remains within the design pressure of 45 psig, including an initial positive pressure of 1.6 psig.

RCS Loops – Modes 3, 4, and 5 (4.4.1.2.2, 4.4.1.3.2, 3.4.1.4.1.b, 4.4.1.4.1.1)

The minimum narrow range level for heat sink adequacy for Modes 3, 4, and 5 changes from 10% to 30% to be consistent with the required level for the RSGs. During Modes 4 and 5, the wide range channels shall be used unless inoperable and the minimum wide range level is 74%. The surveillance requirement limits include uncertainty and ensure that the SG water level is above the top of the tubes, thereby maintaining an adequate heat sink.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (10 CFR 50.92)

Special Test Exceptions – RC Loops (3.10.4)

The special test exception permits hot rod drop time testing during Mode 3 with less than all three RC pumps operating. This exception and associated surveillance requirement is not needed at HNP because rod drop testing is performed only with all three RC pumps operating to simulate a reactor trip under actual conditions.

Borated Water Source – Shutdown (3.1.2.5, B 3/4.1.2)

Based on analysis results, the minimum usable volume during shutdown specified for the boric acid tank in Technical Specification 3.1.2.5 is increased from 6650 to 7150 gallons (associated level changes from 21 to 23%). This change accounts for the increased RCS volume due to the RSGs and other analysis assumptions.

BASIS

This change does not involve a significant hazards consideration for the following reasons:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not affect accident initiators. The change in RCS volume, an input in certain accident analyses, is acceptable since applicable acceptance criteria continue to be met. The change to the calculated residual heat removal (RHR) cooldown time does not adversely impact safe shutdown and meets Standard Review Plan (SRP) requirements. The analysis results for peak accident pressure (P_a) remain below the containment design pressure limit. Revised Technical Specification limits, such as SG minimum water levels, have been analytically determined based on the new SG design features to preserve analysis assumptions. Since the proposed change has no effect on accident initiators, the probability of design basis events is not impacted by these analyses.

   The leak rate assumed in accident analyses is conservative and independent of the value of P_a. Since the peak calculated accident pressure remains below the design value, the containment function as a barrier to the release of radioactive material is not adversely affected. Dose consequences have been analyzed with respect to the proposed change and applicable acceptance criteria continue to be met. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not adversely impact methods of plant operation and new methods of plant systems operation are not created. The impacts of SG replacement on

NSSS and other plant systems, including the change in RCS volume and other design differences, have been evaluated and design and performance requirements continue to be met. Accordingly, no new accident scenarios, failure mechanisms or limiting single failures are introduced.

Changing P_a to reflect the results of current analyses while maintaining its value below the containment design pressure will not introduce significant or adverse changes to the plant design basis that would create a new or different kind of accident.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

Analyses supporting the change reflect the RSG design, including changes to primary and secondary volume and minimum water level for adequate heat sink. Mass and energy and containment response analyses model ESF systems and safety analysis limits. The analysis results demonstrate that applicable acceptance criteria, including DNB, RCS pressure, peak cladding temperature, containment pressure and temperature, and dose, continue to be met.

The calculated containment peak accident pressures result in a value of P_a that remains below the containment design pressure. The margin of safety during design basis events is preserved if P_a does not exceed the design pressure. In addition, performance of the containment leakage rate testing program ensures the validity of the assumptions used in the accident analyses.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (10 CFR 50.92)

Group II Core Limits and RTS/ESFAS

TS #	Title	Description
Fig. 2.1-1 B 2.1.1 3.2.5.a B 3/4.2.5 6.9.1.6	Reactor Core Safety Limits DNB Parameters Core Operating Limits Report	Core safety limits changed based on new core design analysis for Siemens-only fuel T_{avg} maximum limit of 594.8°F Updated referenced analytical methods
Table 2.2-1 Table 3.3-4	RTS Instrument Trip Setpoints ESFAS Instrument Trip Setpoints	SG level, OTΔT, and OPΔT trip setpoints, allowable values, and associated terms changed due to RSG design differences and analysis results

Reactor Core Safety Limits and Core Operating Limits Report (Fig. 2.1-1, B 2.1.1, 6.9.1.6)

Core Safety Limits changed based on analysis for Siemens-only core using the approved Siemens HTP correlation. The revised limits are based on uprated reactor core power of 2900 MWt but are also appropriate (and conservative) for reactor core power of 2775 MWt. The referenced analytical methods for the COLR are updated.

DNB Parameters (3.2.5.a, B 3/4.2.5)

T_{avg}, reduced from the original HNP T_{avg} of 588.8°F to accommodate OSG degradation, will be restored to the original value following SG replacement. The value of T_{avg}, with an appropriate uncertainty value, is used as an input to the transient and accident analyses performed in support of replacement. The results demonstrate that the proposed T_{avg} limit of 594.8 is adequate to maintain a minimum DNB Ratio (DNBR) at or above the design DNBR value during normal operation and analyzed transients.

RTS and ESFAS Instrument Trip Setpoints (Table 2.2-1, Table 3.3-4)

The proposed changes to RTS and ESFAS trip setpoints, allowable values, and associated terms result from design differences between the RSG and OSG and the analyses performed to support operation with the RSG. The analyses incorporate application of Westinghouse margin improvement methodology for OTΔT and OPΔT (approved for use at Farley Nuclear Plant), updated uncertainty calculations, and increases to safety analysis limits. The results of accident analyses demonstrate that, since applicable acceptance criteria continue to be met, the proposed table values are acceptable. The setpoint uncertainty calculations confirm that the setpoints are appropriate for operation with the RSGs at reactor core power of 2775 MWt (or 2900 MWt) and T_{avg} of 588.8°F.

TS Table 2.2-1 values are revised as indicated (shaded items):

Table 2.2-1	Proposed TS Changes (shaded)				
Functional Item	Total Allowance (TA)	Z Term	Sensor Error (S)	Trip Setpoint (TS)	Allowable Value (AV)
3. Power Range, Neutron Flux High Positive Rate	2.5	0.83	0.0	≤ 5% of RTP with a time constant of ≥ 2 seconds	≤ 6.3% of RTP with a time constant of ≥ 2 seconds
4. Power Range, Neutron Flux High Negative Rate	2.5	0.83	0.0	≤ 5% of RTP with a time constant of ≥ 2 seconds	≤ 6.3% of RTP with a time constant of ≥ 2 seconds
7. Overtemperature ΔT	9.0	7.31	Note 5	Note 1	Note 2
8. Overpower ΔT	4.0	2.32	1.3	Note 3	Note 4
9. Pressurizer Pressure – Low, Reactor Trip	5.0	1.52	1.5	≥ 1960 psig	≥ 1948 psig
10. Pressurizer Pressure - High, Reactor Trip	7.5	1.52	1.5	≤ 2385 psig	≤ 2397 psig
11. Pressurizer Water Level - High	8.0	3.42	1.75	≤ 92% of instrument span	≤ 93.5% of instrument span
12. Reactor Coolant Flow - Low	4.58	1.98	0.6	≥ 90.5% of loop full indicated flow	≥ 89.5% of loop full indicated flow
13. SG Water Level, Low-Low	25.0	16.85	2.0	≥ 25% of narrow range instrument span	≥ 23.5% of narrow range instrument span
14. Steam Generator Water Level – Low Coincident w/	8.9	5.35	2.0	≥ 25% of narrow range instrument span	≥ 23.5% of narrow range instrument span
Steam / Feedwater Flow Mismatch	20.0	3.01	Note 6	≤ 40% of full steam flow at RTP	≤ 43.1% of full steam flow at RTP

Table 2.2-1 Table Notations are revised as follows:

Note 1: Overtemperature ΔT

Parameter	Proposed change
K_1	1.185
K_3	0.0012/psig
τ_1	0 sec
τ_2	0 sec
τ_3	4 sec
τ_4	22 sec
T′	= Reference T_{avg} at Rated Thermal Power (≤ 588.8°F)
f_1 (ΔI) Item (2)	1.75%
f_1 (ΔI) Item (3)	1.50%

Note 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input; 2.0% of ΔT span for T_{avg} input; 0.4% of ΔT span for pressurizer pressure input; and 0.7% of ΔT span for ΔI input.

Note 3: Overpower ΔT:

Parameter	Proposed change
K_4	1.12
τ_7	13 sec
T''	Reference T_{avg} at Rated Thermal Power ($\leq$ 588.8°F)

Note 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input and 0.2% of ΔT span for T_{avg} input.

Note 5: The sensor error is 1.3% of ΔT span for ΔT/ T_{avg} temperature measurements; and 1.0% of ΔT span for pressurizer pressure measurements.

Note 6: The sensor error (in % Span of Steam Flow) is: 1.1% for steam flow; 1.8% for feedwater flow; and 2.4% for steam pressure.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (10 CFR 50.92)

Technical Specification Table 3.3-4 values are revised as indicated (shaded items):

Table 3.3-4	Proposed TS Changes (shaded)				
Functional Item	Total Allowance (TA)	Z Term	Sensor Error (S)	Trip Setpoint (TS)	Allowable Value (AV)
1.c Containment Pressure High-1	3.64	0.71	1.5	≤ 3.0 psig	≤ 3.6 psig
1.d Pressurizer Pressure - Low, Safety Injection	18.75	10.47	1.5	≥ 1850 psig	≥ 1838 psig
1.e Steamline Pressure - Low	4.52	0.71	2.0	≥ 601 psig	≥ 581.5 psig
2.c Containment Pressure High-3	3.64	0.71	1.5	≤ 10.0 psig	≤ 11.0 psig
4.c Containment Pressure High-2	3.64	0.71	1.5	≤ 3.0 psig	≤ 3.6 psig
4.e Negative Steamline Pressure Rate – High	2.3	0.5	0.0	≤ 100 psi	≤ 119.5 psi
5.b SG Water Level - High-High	22.0	9.63	2.0	≤ 78.0% of narrow range	≤ 79.5% of narrow range instrument
6.c SG Water Level, Low-Low	25.0	16.85	2.0	≥ 25.0% of narrow range instrument span	≥ 23.5% of narrow range instrument span
6.g Steamline Differential Pressure – High	5.0	0.87	3.0	≤ 100 psi	≤ 127.4 psi
10.a Pressurizer Pressure P-11	N/A	N/A	N/A	≥ 2000 psig	≥ 1988 psig
NOT P-11	N/A	N/A	N/A	≤ 2000 psig	≤ 2012 psig

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (10 CFR 50.92)

BASIS

This change does not involve a significant hazards consideration for the following reasons:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   HNP structures, systems and components, as well as the RSGs, are designed to operate at the original T_{avg} value of 588.8°F. The comprehensive engineering review performed to support SG replacement includes evaluation and analysis results for transients and design basis accidents that meet applicable acceptance criteria. The proposed DNB parameter values ensure the core and reactor coolant system do not exceed their safety limits. The trip setpoints initiate safeguards actions that mitigate postulated accident consequences. Since plant systems will function as designed and performance requirements are verified to be acceptable with the proposed changes, there are no impacts to accident initiators or precursors. Therefore, these changes do not involve a significant increase in the probability of an accident previously evaluated.

   The RSGs are designed to minimize potential for the types of tube degradation experienced with the OSGs. The changes to reactor core safety limits, RTS/ESFAS values and T_{avg} limit support transient and accident analysis assumptions relating to fuel clad failure. Since SG tube and fuel clad integrity will continue to be adequately maintained, these barriers to release of radioactive material will perform their required function. All dose consequences continue to meet acceptance criteria. As a result, the changes do not involve a significant increase in consequences of previously evaluated accidents.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   Since HNP structures, systems and components are designed for operation at the proposed T_{avg}, returning to this value following SG replacement does not introduce a new method of plant operation. The proposed reactor core safety limits, RTS/ESFAS and T_{avg} limit changes do not introduce new failure mechanisms or limiting single failures since the values have been shown to preserve transient and accident analysis assumptions. The acceptable evaluation results demonstrate that the changes do not adversely affect or challenge the performance or integrity of safety related systems. Hence, no new accident scenarios are created.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

   All evaluations and analyses performed to support SG replacement reflect the proposed values. The results demonstrate that applicable acceptance criteria (including DNB, RCS pressure and temperature, LOCA peak clad temperature, containment pressure and temperature, and dose) continue to be met. Calculations confirm that the proposed RTS/ESFAS values provide margin to the safety analysis limits, thereby ensuring that the margin of safety inherent in the safety analysis limit itself is preserved.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (10 CFR 50.92)

Group III Dose and Specific Activity

TS #	Title	Description
3.4.8 Fig. 3.4-1 Table 4.4-4 B 3/4.4.8	Specific Activity	DEI-131 limit 0.35 μCi/gm for 48 hr interval; 60 μCi/gm for all power levels for single sample.
1.11	Definitions – Dose Equivalent I-131	Defines basis for Dose Equivalent I-131

Specific Activity (3.4.8)

The limit for specific activity is reduced from 1.0 μCi/gm to 0.35 μCi/gm DEI131 based on steam generator tube rupture (SGTR) analysis results. The change conservatively limits the quantity of radioactive material available for release in the event of SGTR and other design basis accidents by reducing the source term in the RCS inventory. The proposed limit assures that dose acceptance criteria are met for all event analyses. The limit of 60 μCi/gm DEI-131 at any power level for a single RCS sample is conservative, since it replaces Figure 3.4-1 that allowed increasing specific activity levels for power levels below 80%.

Emergency operating procedure (EOP) steps/sequence are changed to be consistent with current Westinghouse Owners Group emergency response guidelines and to satisfy SGTR analysis assumptions for margin to overfill. The procedure changes have been demonstrated to be acceptable by shift crew simulator exercises. There are no additional operator qualifications, support personnel and/or equipment needed as a result of these changes. supporting accident analyses incorporate updated and conservative assumptions and methodologies.

Definitions – Dose Equivalent I-131 (1.11)

The basis for the thyroid dose conversion factors used in the calculation of Dose Equivalent I-131 is updated from that used in the analysis of record in order to be consistent with current analysis techniques and NRC expectations. ICRP-30 (or the essentially equivalent FGR No. 11) is utilized for the SGR/PUR analyses.

BASIS

This change does not involve a significant hazards consideration for the following reasons:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change reduces Technical Specification limits on reactor coolant activity to ensure that the dose consequences of the proposed new SGTR analysis meet guidelines. The Technical Specification limits are not accident initiators and the proposed change provides additional conservatism in the limiting initial condition for RCS activity level. The change in these limits functions only to mitigate consequences of RCS liquid and steam release events and does not impact methods for plant operation. Since there are no

adverse impacts to initiators of accidents previously evaluated, there is no significant increase in the probability of such accidents.

The proposed change to the basis for thyroid dose conversion factors is an accepted update to that used in the analysis of record. Dose consequences for all analyses, including SGTR, remain within allowable guidelines. Although there are increases to previously reported dose consequences, the increase is not significant since the results remain under the SRP acceptance criteria.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to specific activity limits does not modify plant systems, structures, or components or impact methods of plant operation. The SGTR analysis that provides the basis for the change demonstrates adequate margin to overfill based on revised emergency operating procedures. Operator actions and response times have been demonstrated to be acceptable and to satisfy analysis assumptions in simulator exercises. The operator actions do not represent different types of actions from those currently included in the EOP's, and are not subject to errors that would create new failure mechanisms.

No new types of failures and no new or different accident initiators are created. As a result, the change will not introduce significant or adverse changes to the plant design basis that could lead to the creation of a new or different kind of accident.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

The specific activity limit has no effect on safety limits or limiting safety system settings. The only impact of the specific activity limits is on dose consequences for certain design basis accidents such as SGTR. The proposed change is in the conservative direction for dose. The SGTR analysis is the most limiting of these accident dose evaluations. The proposed new analysis demonstrates that the SGTR consequences remain within allowable SRP guideline limits, which establish the margin of safety. Since the analysis results remain below the SRP guidelines, the margin of safety is not reduced.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (10 CFR 50.92)

Group IV Plant Systems & Other

TS #	Title	Description
Table 3.7-1	Max Power Range Neutron Flux High Setpoint w/ Inoperable MSSVs	Lower setpoint values based on analysis for uprate power – conservative for existing power
B 3/4.7.1.3	Condensate Storage Tank	6 hrs at Hot Standby compatible with existing volume
3/4.4.5 B 3/4.4.5	SG Surveillance Requirements	Eliminates F* criteria and option to sleeve tubes; provides one-time exemption for in-service testing
B 3/4.7.2	SG Pressure/Temperature Limits	Clarifies basis and indicates RSG RT_{NDT}
B 3/4.7.1.2	Auxiliary Feedwater System	Change methodology for maximum allowable degradation of pump performance

Maximum Power Range Neutron Flux High Setpoint with Inoperable Main Steam Safety Valves (Technical Specification Figure B 3/4.4-1)

The setpoints are inversely proportional to power level and are reduced to accommodate future power uprate. The proposed values are conservative for operation with the RSGs at current power level.

Condensate Storage Tank (Technical Specification B 3.7.1.3)

Reducing specified time at hot standby for plant cooldown from 12 hours to 6 hours avoids the need to increase minimum stored CST volume. The analysis considers the increases in RCS volume and sensible heat due to the RSGs and conservatively assumes operation at 2900 MWt. The 6 hour period exceeds the SRP minimum requirement of 4 hours.

Steam Generator Surveillance Requirements (Technical Specification 3/4.4.5)

Surveillance and repair requirements that were introduced due to degradation problems associated with the OSG and other early model Steam generators are not applicable to the RSGs as a result of design differences. The text that relates to F* criteria, sleeving, and inspections of sleeves and preheater region tubes is therefore deleted.

Auxiliary Feedwater System (Technical Specification B 3/4.7.1.2)

The methodology for determining minimum allowable AFW pump performance was changed from a percentage of the vendor pump curve to a maximum allowable degradation. Existing surveillance minimum requirements are maintained.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (10 CFR 50.92)

BASIS

This change does not involve a significant hazards consideration for the following reasons:

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

1. Reducing the required time at hot standby to maintain adequate CST minimum volume ensures accident mitigation capability and is not involved with accident initiation. Conservative reactor trip setpoints during power operation with inoperable main steam line safety valves ensure accident analysis assumptions for maximum secondary system pressure. The proposed setpoint changes are also unrelated to initiation of previously evaluated accidents.

   The removal of F* plugging criteria is consistent with NUREG-0452, "Standard Technical Specifications." The remaining Technical Specification plugging criterion is based on a minimum wall thickness due to wastage as determined by ASME Section XI. The proposed change is conservative because operation with F* degraded tubes is eliminated.

   The potential for a tube rupture is expected to be acceptably low based on the qualification analysis and testing for the RSGs. The proposed program for periodic in-service inspection monitors the integrity of the SG tubing to provide reasonable assurance that there is sufficient time to take proper and timely corrective action if any tube degradation is detected. The deleted OSG tube inspection requirements are not applicable to the RSG due to design differences.

   SG tube inspections are not accident initiators and the function of the inspection program is to identify and eliminate, when appropriate, accident precursor conditions. Because of this, the probability of postulated accidents (eg, SG tube rupture) is not increased.

   Since the proposed changes are either conservative (reduced RTS setpoints, elimination of F* tube plugging criteria) or related to mitigation of postulated accidents (hot standby time prior to cooldown), there is no significant increase in dose consequences.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The revised thermal power restrictions imposed by Technical Specification when there are inoperable main steam line safety valves are conservative and consistent with the Technical Specification Bases methodology. The reduction in hot standby time to six hours continues

to satisfy requirements for safe shutdown. No new failure mechanisms or limiting single failures are introduced and there are no adverse effects or challenges to the performance or integrity of safety related systems.

Removing application of F* repair criteria and sleeving methodologies from Technical Specification with the RSGs does not introduce significant or adverse changes to the plant design basis that could lead to a new or different kind of accident being created. These requirements were needed to address design and material conditions of the original HNP SGs that are not applicable to the RSGs. This change does not alter the objective of SG surveillance activities - maintaining the structural integrity of this portion of the reactor coolant system. The surveillance activities are performed during outages and the proposed surveillance program is consistent with regulatory guidance. No new failures are created.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

The thermal power restrictions imposed by Technical Specification when there are inoperable main steam line safety valves ensures sufficient relieving capacity during all postulated transients, thereby maintaining margin of safety for secondary system pressure.

The proposed in-service inspection program for the RSGs monitors the integrity of the SG tubing and is consistent with NUREG-0452. Since the F* repair criteria and sleeving methods specifically addressed OSG degradation issues, their removal does not involve a reduction in a margin of safety. Analyses demonstrate that the RSG tubing retains adequate structural and leakage integrity using the proposed plugging criteria during normal, transient, and postulated accident conditions. The RSG tubing design and proposed plugging criteria are consistent with applicable ASME requirements and do not allow for operation with indications identified by F* criteria.

The proposed plugging limit maintains the margin of safety by providing for leakage detection and shutdown in the event of an unexpected tube leak while minimizing the potential for excessive leakage or tube burst in the event of a MSLB or LOCA.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63

TECHNICAL SPECIFICATION CHANGE REQUEST
STEAM GENERATOR REPLACEMENT

ENVIRONMENTAL EVALUATION (10 CFR 51.22)

10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; (3) result in a significant increase in individual or cumulative occupational radiation exposure. Carolina Power & Light Company (CP&L) has reviewed this amendment request and determined the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental assessment or environmental impact statement needs to be prepared in connection with the issuance of the amendment. The basis for this determination is as follows:

CP&L has completed a comprehensive engineering review program in support of steam generator replacement (SGR) at the Harris Nuclear Plant (HNP). CP&L plans to replace the original Westinghouse Model D4 steam generators (OSG) with Westinghouse Model Delta 75 replacement steam generators (RSG). The program considers the OSG - RSG differences in conjunction with restoration of the original nominal reactor coolant average temperature (T_{avg}) of 588.8 °F. While the results of the supporting analyses and evaluations support plant operation with the RSG at both the current licensed core power of 2775 MWt and an uprated core power level of 2900 MWt, this amendment application does not include a request for a power uprate.

CP&L's environmental impact evaluation of the proposed steam generator replacement also considers the power uprate option. The environmental impact evaluation is included as Section 2.26 of the BOP Licensing Report (Enclosure 7) and is based upon the assessment of environmental impact reported in the Final Environmental Statement related to the operation of Shearon Harris Nuclear Power Plant, Units 1 and 2, prepared by the U.S. Nuclear Regulatory Commission (NRC) in October 1983 (FES-OL). The environmental evaluation for steam generator replacement with the power uprate option addresses the environmental impacts addressed in the FES-OL, including impacts on cooling power drift, thermal discharges to the ultimate heat sink (i.e., the Auxiliary Reservoir), non-radiological effluents, noise impacts, and terrestrial impacts. In the FES-OL, the NRC staff determined "...The Shearon Harris Station can be operated with minimal environmental impact." The environmental impacts associated with steam generator replacement and power uprate are bounded by the impacts reported in the FES-OL for the 2 unit site. It is concluded, therefore, that there is no significant change in the types of significant increase in the amounts of any effluents that may be released off site.

Additionally, compared to the original Model D4 steam generators, the Model Delta 75 RSGs include enhanced maintenance and access provisions such as ladders, grab bars, and manways for inspection of the steam generator upper internals. Enhanced maintenance and access provisions in conjunction with channel head and divider plate electropolishing and low cobalt Inconel material selection are provided to reduce radiation exposure. The manways on the HNP Model Delta 75 RSGs utilize studs, nuts and washers for securing the closure covers. This permits use of stud tensioners for closure assembly and disassembly, thus reducing radiation exposure. The original Model D4 steam generator manway closures used bolted fasteners. Additionally, the thermally treated alloy 690 tube material used in the Model Delta 75 RSGs is considered to be a material enhancement which has been proven through laboratory testing and operational experience to provide increased corrosion resistance compared to the mill annealed Alloy 600, the Model D4 tube material, particularly with regard to pressurized water stress corrosion cracking (PWSSC). As such, Alloy 690 has been determined to be the material of choice for steam generator applications. Based on these considerations, it can be concluded that the replacement steam generators will not result in a significant increase in individual or cumulative occupational exposures.

In summary, the replacement of the steam generators at HNP as described herein involves no significant hazards consideration (see Enclosure 2), does not significantly change the types or significantly increase the amounts of any effluents that may be released offsite, or significantly increase individual or cumulative occupational radiation exposure. The replacement activities do not constitute any other type of new and appreciable environmental impact. It is concluded, therefore, that the steam generator replacement activities are essentially benign with respect to the human environment.

Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental assessment or environmental impact statement is required.

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
TECHNICAL SPECIFICATION CHANGE REQUEST
STEAM GENERATOR REPLACEMENT

PAGE CHANGE INSTRUCTIONS

Remove Pages:	Insert Pages:
Index page vii	Index page vii
Index page xiii	Index page xiii
1-2a	1-2a
2-2	2-2
2-4	2-4
2-5	2-5
2-7	2-7
2-8	2-8
2-9	2-9
2-10	2-10
B 2-1	B 2-1
B 2-6	B 2-6
3/4 1-11	3/4 1-11
3/4 2-14	3/4 2-14
3/4 3-28	3/4 3-28
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32
3/4 3-33	3/4 3-33
3/4 3-34	3/4 3-34
3/4 4-2	3/4 4-2
3/4 4-3	3/4 4-3
3/4 4-5	3/4 4-5
3/4 4-6	3/4 4-6
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
TECHNICAL SPECIFICATION CHANGE REQUEST
STEAM GENERATOR REPLACEMENT

PAGE CHANGE INSTRUCTIONS (cont.)

Remove Pages:	Insert Pages:
3/4 4-15	3/4 4-15
3/4 4-16	3/4 4-16
3/4 4-16a	No Replacement
3/4 4-17	3/4 4-17
3/4 4-19	3/4 4-19
3/4 4-20	3/4 4-20
3/4 4-20a	3/4 4-20a
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
3/4 4-31	3/4 4-31
3/4 7-2	3/4 7-2
3/4 10-4	3/4 10-4
B 3/4 1-3	B 3/4 1-3
B 3/4 2-6	B 3/4 2-6
B 3/4 4-1	B 3/4 4-1
B 3/4 4-2 *	B 3/4 4-2 *
B 3/4 4-3	B 3/4 4-3
B 3/4 4-5	B 3/4 4-5
B 3/4 6-1a	B 3/4 6-1a
B 3/4 6-2	B 3/4 6-2
B 3/4 7-2	B 3/4 7-2
5-6	5-6
6-19c	6-19c
6-24a	6-24a
6-24b	6-24b
6-24c	6-24c

* Page has changed only due to information rolling over from previous page

Enclosure 5 to SERIAL: HNP-00-142

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
TECHNICAL SPECIFICATION CHANGE REQUEST
STEAM GENERATOR REPLACEMENT

PROPOSED TECHNICAL SPECIFICATION PAGE CHANGES

ORIGINAL MARKED UP PAGES

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

Startup and Power Operation 3/4 4-1

Hot Standby 3/4 4-2

Hot Shutdown 3/4 4-4

Cold Shutdown - Loops Filled 3/4 4-6

Cold Shutdown - Loops Not Filled 3/4 4-7

3/4.4.2 SAFETY VALVES

Shutdown 3/4 4-8

Operating 3/4 4-9

3/4.4.3 PRESSURIZER 3/4 4-10

3/4.4.4 RELIEF VALVES 3/4 4-11

3/4.4.5 STEAM GENERATORS 3/4 4-13

TABLE 4.4-1 MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION 3/4 4-18

Delete

TABLE 4.4-2A STEAM GENERATOR TUBE INSPECTION 3/4 4-19

Delete

TABLE 4.4-2B STEAM GENERATOR TUBE INSPECTION - TUBE EXPANDED IN PREHEATER REGION 3/4 4-20

(DELETED) Add

Add

TABLE 4.4-2C DELETED 3/4 4-20a

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

Leakage Detection Systems 3/4 4-21

Operational Leakage 3/4 4-23

TABLE 3.4-1 REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES . . . 3/4 4-25

3/4.4.7 CHEMISTRY 3/4 4-26

TABLE 3.4-2 REACTOR COOLANT SYSTEM CHEMISTRY LIMITS 3/4 4-27

TABLE 4.4-3 REACTOR COOLANT SYSTEM CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS 3/4 4-28

3/4.4.8 SPECIFIC ACTIVITY 3/4 4-29

Add

Delete

FIGURE 3.4-1 DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 μCi/gram DOSE EQUIVALENT I-131 . . . . . 3/4 4-30

(DELETED)

TABLE 4.4-4 REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM 3/4 4-31

INDEX

3.0/4.0 BASES

SECTION	PAGE
3/4.0 APPLICABILITY	B 3/4 0-1
3/4.1 REACTIVITY CONTROL SYSTEMS	
3/4.1.1 BORATION CONTROL	B 3/4 1-1
3/4.1.2 BORATION SYSTEMS	B 3/4 1-2
3/4.1.3 MOVABLE CONTROL ASSEMBLIES	B 3/4 1-3
3/4.2 POWER DISTRIBUTION LIMITS	B 3/4 2-1
3/4.2.1 AXIAL FLUX DIFFERENCE	B 3/4 2-1
3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, AND RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR	B 3/4 2-2
FIGURE B 3/4.2-1 (DELETED)	B 3/4 2-3
3/4.2.4 QUADRANT POWER TILT RATIO	B 3/4 2-5 -6
3/4.2.5 DNB PARAMETERS	B 3/4 2-6
3/4.3 INSTRUMENTATION	
3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION	B 3/4 3-1
3/4.3.3 MONITORING INSTRUMENTATION	B 3/4 3-3
3/4.3.4 (DELETED)	B 3/4 3-6
3/4.4 REACTOR COOLANT SYSTEM	
3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION	B 3/4 4-1
3/4.4.2 SAFETY VALVES	B 3/4 4-1
3/4.4.3 PRESSURIZER	B 3/4 4-2
3/4.4.4 RELIEF VALVES	B 3/4 4-2
3/4.4.5 STEAM GENERATORS	B 3/4 4-2 6
3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE	B 3/4 4-3
3/4.4.7 CHEMISTRY	B 3/4 4-4
3/4.4.8 SPECIFIC ACTIVITY	B 3/4 4-5
3/4.4.9 PRESSURE/TEMPERATURE LIMITS	B 3/4 4-6

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

Delete

Insert

International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, Volume 3 No. 1-4, 1979 (or equivalently, Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", EPA 520/1-88-020, September 1988).

Delete

Replace with next page

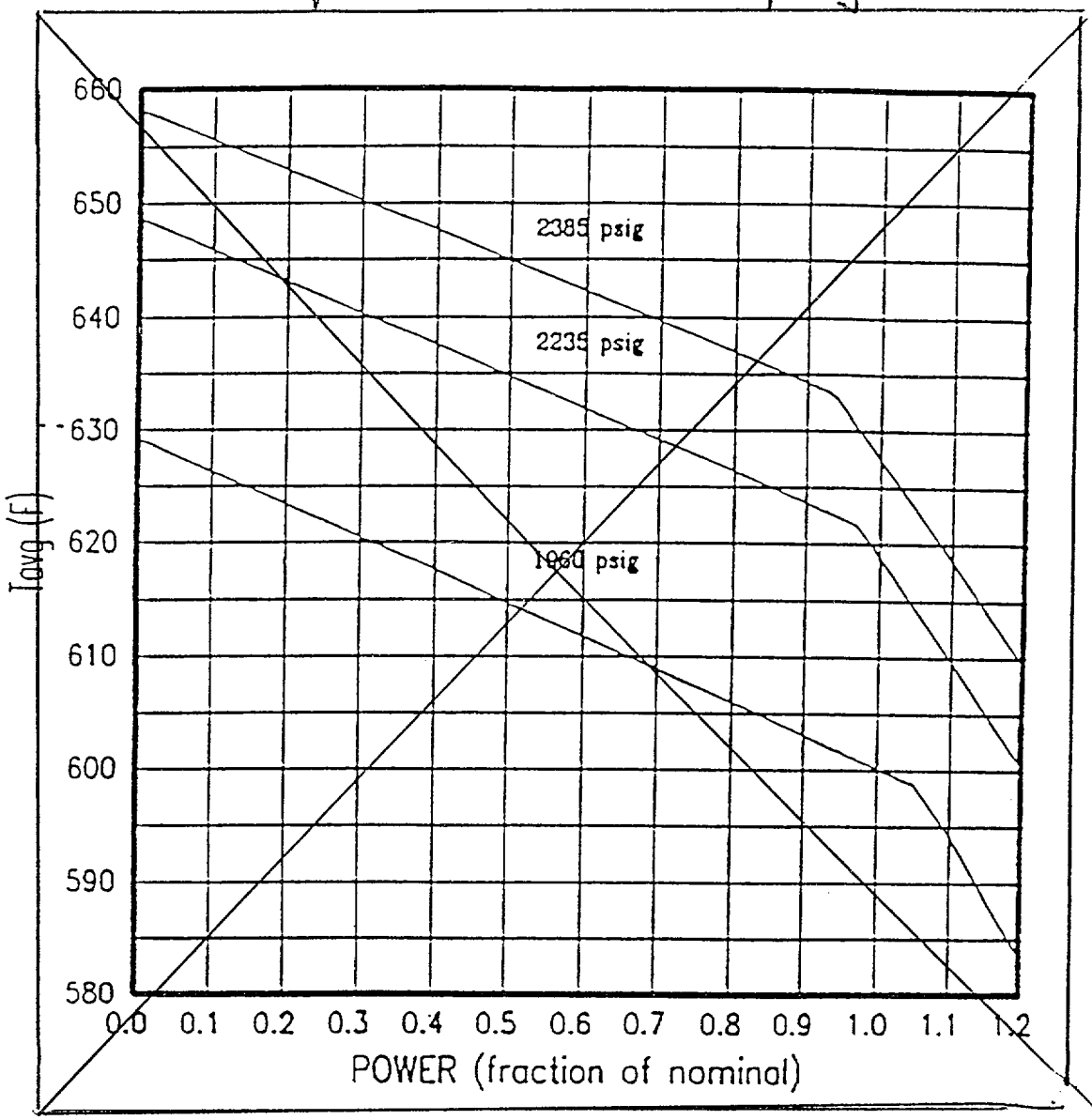
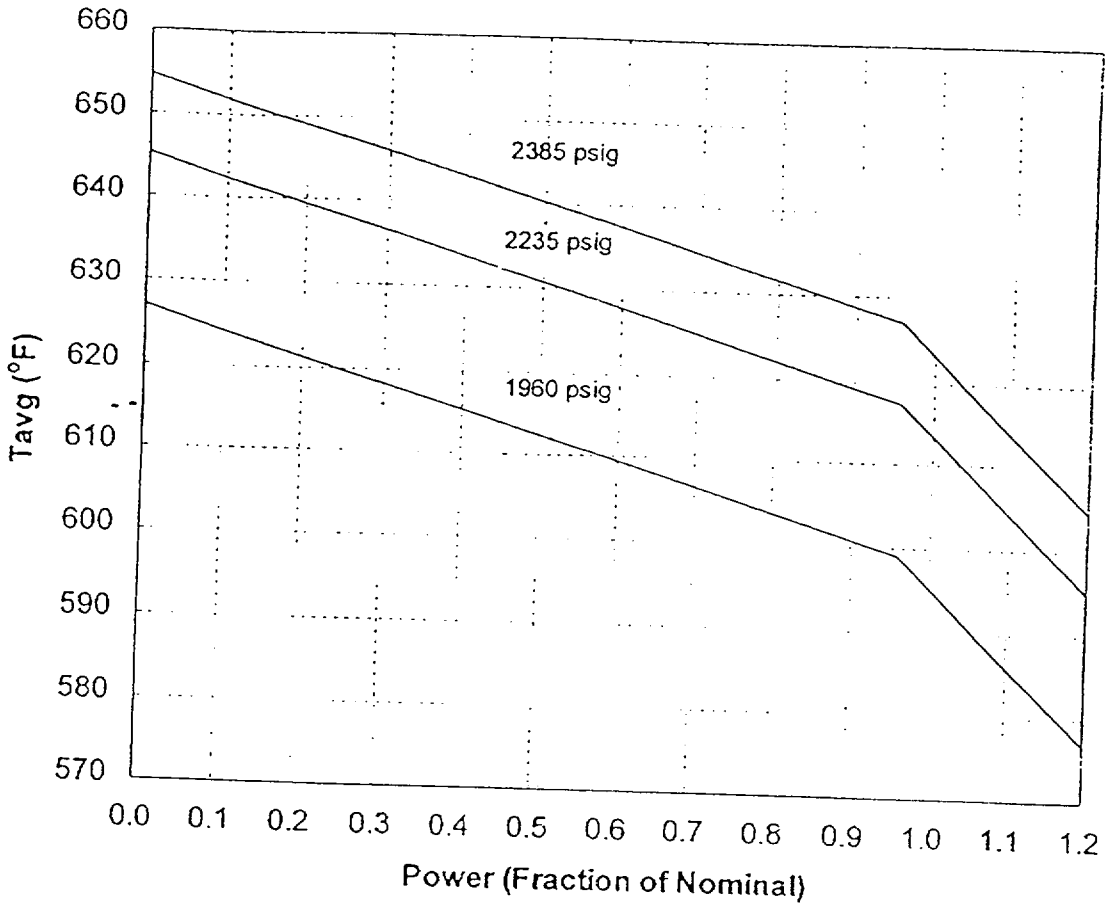


FIGURE 2.1-1
REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION
WITH MEASURED RCS FLOW ≥ [293,540 GPM X (1.0 + C_1)]

Delete



Replacement Figure 2.1-1

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤ 109% of RTP**	≤ 111.1% of RTP**
b. Low Setpoint	8.3	4.56	0	≤ 25% of RTP**	≤ 27.1% of RTP**
3. Power Range, Neutron Flux, High Positive Rate	~~1.6~~ 2.5	~~0.5~~ 0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	~~1.6~~ 2.5	~~0.5~~ 0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% of RTP**	≤ 30.9% of RTP**
6. Source Range, Neutron Flux	17.0	10.01	0	≤ 10^5 cps	≤ 1.4 x 10^5 cps
7. Overtemperature ΔT	~~8.7~~ 9.0	~~6.02~~ 7.31	Note 5	See Note 1	See Note 2
8. Overpower ΔT	~~4.7~~ 4.0	~~1.50~~ 2.32	~~1.9~~ 1.3	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	~~2.21~~ 1.52	1.5	≥ 1960 psig	≥ ~~1946~~ 1948 psig
10. Pressurizer Pressure-High	7.5	~~5.01~~ 1.52	~~0.5~~ 1.5	≤ 2385 psig	≤ ~~2399~~ 2397 psig
11. Pressurizer Water Level-High	8.0	~~2.18~~ 3.42	~~1.5~~ 1.75	≤ 92% of instrument span	≤ ~~93.0~~ 93.5% of instrument span

**RTP = RATED THERMAL POWER

Delete

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	~~2.9~~ 4.58	1.98	0.6	≥ 90.5% of loop full indicated flow	≥ 89.5% of loop full indicated flow
13. Steam Generator Water Level Low-Low	~~19.2~~ 25.0	~~14.06~~ 16.85	~~2.97~~ 2.0	≥ ~~38.5~~ 25.0% of narrow range instrument span	≥ ~~36.5~~ 23.5% of narrow range instrument span
14. Steam Generator Water Level - Low Coincident With Steam/Feedwater Flow Mismatch	~~19.2~~ 8.9	~~2.23~~ 5.35	~~2.97~~ 2.0	≥ ~~38.5~~ 25.0% of narrow range instrument span	≥ ~~36.5~~ 23.5% of narrow range instrument span
	20.0	~~3.41~~ 3.01	Note 6	≤ 40% of full steam flow at RTP**	≤ 43.1% of full steam flow at RTP**
15. Undervoltage - Reactor Coolant Pumps	14.0	1.3	0.0	≥ 5148 volts	≥ 4920 volts
16. Underfrequency - Reactor Coolant Pumps	5.0	3.0	0.0	≥ 57.5 Hz	≥ 57.3 Hz
17. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥ 1000 psig	≥ 950 psig
b. Turbine Throttle Valve Closure	N.A.	N.A.	N.A.	≥ 1% open	≥ 1% open
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

**RTP = RATED THERMAL POWER

Delete

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left[\frac{1}{1 + \tau_3 S} \right] \leq \Delta T_o \left\{ K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left[\frac{1}{1 + \tau_6 S} \right] - T' \right] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where:

ΔT	=	Measured ΔT by RTD Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = ~~8~~ 0 s, τ_2 = ~~3~~ 0 s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constants utilized in the lag compensator for ΔT, τ_3 = ~~0~~ 4 s;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	~~1.17~~; 1.185
K_2	=	0.0224/°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = ~~20~~ 22 s, τ_5 = 4 s;

Delete

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: (Continued)

T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;
T'	= ≤	~~580.8°F (Nominal~~ T_{avg} at RATED THERMAL POWER~~)~~; Reference (≤ 588.8°F);
K_3	=	~~0.001072~~/psig; 0.0012
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between -21.6% and +12.0%, f_1 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -21.6%, the ΔT Trip Setpoint shall be automatically reduced by ~~2.36~~% of its value at RATED THERMAL POWER; and 1.75

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 12.0%, the ΔT Trip Setpoint shall be automatically reduced by ~~1.57~~% of its value at RATED THERMAL POWER. 1.50

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~2.1% ΔT span.~~ 1.4% of ΔT span for ΔT input; 2.0% of ΔT span for Tavg input; 0.4% of ΔT span for pressurizer pressure input; and 0.7% of ΔT span for ΔI input.

Delete

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1+\tau_7 S}\right) \left(\frac{1}{1+\tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1+\tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:

ΔT	=	As defined in Note 1,
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	As defined in Note 1,
τ_1, τ_2	=	As defined in Note 1,
$\frac{1}{1+\tau_3 s}$	=	As defined in Note 1,
τ_3	=	As defined in Note 1,
ΔT_o	=	As defined in Note 1,
K_4	=	~~1.079~~, 1.12
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature,
$\frac{\tau_7 S}{1+\tau_7 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation,
τ_7	=	Time constants utilized in the rate-lag compensator for T_{avg}, τ_7 = ~~10~~ 13 s,
$\frac{1}{1+\tau_6 S}$	=	As defined in Note 1,
τ_6	=	As defined in Note 1,

Debbie

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: (Continued)

K_6 = 0.002/°F for T > T" and K_6 = 0 for T ≤ T",

T = As defined in Note 1,

Reference

T" = ~~Indicated~~ T_{avg} at RATED THERMAL POWER ~~(Calibration temperature for ΔT instrumentation, ≤ 580.8~~°F).

(≤ 588.8

S = As defined in Note 1, and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~2.3% ΔT span~~ **1.4% of ΔT span for ΔT input and 0.2% of ΔT span for Tavg input.**

~~NOTE 5: The sensor error for temperature is 1.9 and 1.1 for pressure.~~

~~NOTE 6: The sensor error for steam flow is 0.9, for feed flow is 1.5, and for steam pressure is 0.75.~~

NOTE 5: The sensor error is: 1.3% of ΔT span for ΔT/Tavg temperature measurements; and 1.0% of ΔT span for pressurizer pressure measurements.

NOTE 6: The sensor error (in % Span of Steam Flow) is: 1.1% for steam flow; 1.8% for feedwater flow; and 2.4% for steam pressure.

Delete

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. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform 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.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (in this application, the HTP correlation for Siemens Fuel ~~and the WRB-1 or WRB-2 correlation for Westinghouse Fuel~~). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

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

Delete

Reactor Coolant Flow (Continued)

(a power level of approximately 49% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90.5% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

the setpoint value.

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by ~~greater than or equal to~~ ~~1.627×10^6 lbs/hour~~. The Steam Generator Low Water level portion of the trip is activated when the ~~water level drops below 38.5%~~, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

the setpoint value is reached

Undervoltage and Underfrequency - Reactor Coolant Pump Buses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint 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 1.2 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 Setpoint is reached shall not exceed 0.3 second.

On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure

Delete

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. The boric acid tank with:

1. A minimum contained borated water volume of ~~6650~~ 7150 gallons which is ensured by maintaining indicated level of greater than or equal to ~~21~~ 23%,

2. A boron concentration of between 7000 and 7750 ppm, and

3. A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1. A minimum contained borated water volume of 106,000 gallons, which is equivalent to 12% indicated level,

2. A boron concentration of between 2400 and 2600 ppm, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume, and

3. Verifying the boric acid tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 40°F.

Delete

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

a. Reactor Coolant System T_{avg} ≤ ~~586.1~~ 594.8°F after addition for instrument uncertainty, and

b. Pressurizer Pressure ≥ 2185 psig* after subtraction for instrument uncertainty, and

c. RCS total flow rate ≥ 293,540 gpm after subtraction for instrument uncertainty.

Delete

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters not within its specified limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 6 hours.

Delete

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown in Specification 3.2.5 shall be verified to be within its limit at least once per 12 hours.

4.2.5.2 Verify, by precision heat balance, that RCS total flow rate is within its limit at least once per 18 months.**

Delete

* This limit is not applicable during either a THERMAL POWER Ramp in excess of ±5% RATED THERMAL POWER per minute or a THERMAL POWER step change in excess of ±10% RATED THERMAL POWER.

** Required to be performed within 24 hours after ≥ 95% RATED THERMAL POWER.

Delete

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-1	~~2.7~~ 3.64	0.71	1.5	≤ 3.0 psig	≤ 3.6 psig
d. Pressurizer Pressure--Low	~~18.8~~ 18.75	~~14.41~~ 10.47	1.5	≥ 1850 psig	≥ ~~1836~~ 1838 psig
e. Steam Line Pressure--Low	~~17.7~~ 4.52	~~14.81~~ 0.71	~~1.5~~ 2.0	≥ 601 psig	≥ ~~570.3~~ 581.5 psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-3	~~3.6~~ 3.64	0.71	1.5	≤ 10.0 psig	≤ 11.0 psig

Amendment No. Add

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation (Continued)					
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
4) Containment Radioactivity					
a) Area Monitors (both preentry and normal purges)	See Table 3.3-6, Item ~~1a~~ 1.a for trip setpoint.				
b) Airborne Gaseous Radioactivity					
(1) RCS Leak Detection (normal purge)	See Table 3.3-6, Item ~~1b1~~ 1.b.1 for trip setpoint.				
(2) Preentry Purge Detector	See Table 3.3-6, Item ~~1b2~~ 1.b.2 for trip setpoint.				
c) Airborne Particulate Radioactivity					
(1) RCS Leak Detection (normal purge)	See Table 3.3-6, Item ~~1c1~~ 1.c.1 for trip setpoint.				
(2) Preentry Purge Detector	See Table 3.3-6, Item ~~1c2~~ 1.c.2 for trip setpoint.				
5) Manual Phase "A" Isolation	N.A.	N.A.	N.A.	N.A.	N.A.

Add Amendment No.

TABLE 3.3-1 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. Main Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-2	~~2.7~~ 3.64	0.71	1.5	≤3.0 psig	≤3.6 psig
d. Steam Line Pressure--Low	See Item 1.e. above for Steam Line Pressure--Low Trip Setpoints and Allowable Values.				
e. Negative Steam Line Pressure Rate--High	2.3	0.5	0	≤ 100 psi#	≤ ~~122.0~~ 119.5 psi#**
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.

Add Amendment No.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation (Continued)					
b. Steam Generator Water Level--High-High (P-14)	~~15.0~~ 22.0	~~11.25~~ 9.63	~~2.97~~ 2.0	≤ ~~82.4~~ 78.0% of narrow range instrument span.	≤ ~~84.3~~ 79.5% of narrow range instrument span.
c. Safety Injection	See Item 1. above for Safety Injection Trip Setpoints and Allowable Values.				
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level--Low-Low	~~19.2~~ 25.0	~~14.06~~ 16.85	~~2.97~~ 2.0	≥ ~~38.5~~ 25.0% of narrow range instrument span.	≥ ~~36.5~~ 23.5% of narrow range instrument span.
d. Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump	See Item 9. below for all Loss-of-Offsite Trip Setpoint and Allowable Values.				
f. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	N.A.	N.A.	N.A.	N.A.	N.A.

Delete

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)					
g. Steam Line Differential Pressure--High	5.0	~~1.47~~ 0.87	3.0	≤ 100psi	≤ 127.4 psi
Coincident With Main Steam Line Isolation (Causes AFW Isolation)	See Item 4. above for Main Steam Line Isolation Trip Setpoints and Allowable Values.				
7. Safety Injection Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level--Low-Low	N.A.	N.A.	N.A.	≥ 23.4%	≥ 20.4%
Coincident With Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
8. Containment Spray Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST--Low-Low	See Item 7.b. above for all RWST--Low-Low Trip Setpoints and Allowable Values.				
Coincident With Containment Spray	See Item 2. above for all Containment Spray Trip Setpoints and Allowable Values.				

Amendment No. Add

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Loss-of-Offsite Power					
a. 6.9 kV Emergency Bus Undervoltage--Primary	N.A.	N.A.	N.A.	≥ 4830 volts with a ≤ 1.0 second time delay.	≥ 4692 volts with a time delay ≤ 1.5 seconds
b. 6.9 kV Emergency Bus Undervoltage--Secondary	N.A.	N.A.	N.A.	≥ 6420 volts with a ≤ 16 second time delay (with Safety Injection).	≥ 6392 volts with a time delay ≤ 18 seconds (with Safety Injection).
				≥ 6420 volts with a ≤ 54.0 second time delay (without Safety Injection).	≥ 6392 volts with a ≤ 60 second time delay (without Safety Injection).
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≥ 2000 psig	≥ ~~1986~~ 1988 psig
Not P-11	N.A.	N.A.	N.A.	≤ 2000 psig	≤ ~~2014~~ 2012 psig
b. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	≥ 553°F	≥ 549.3°F

Delete

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with two reactor coolant pumps in operation when the Reactor Trip System breakers are closed or with one reactor coolant pump in operation when the Reactor Trip System breakers are open:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,

APPLICABILITY: MODE 3.[**] — Delete

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.

c. With no reactor coolant loop in operation, immediately open the Reactor Trip System breakers, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

---

*All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

Delete — ~~**See Special Test Exception 3.10.4.~~

Add

SURVEILLANCE REQUIREMENTS (Continued)

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying narrow range secondary side water level to be greater than or equal to ~~10%~~ 30% at least once per 12 hours.

4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying ~~narrow range secondary side water level to be greater than or equal to 10%~~ at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

wide range (WR) secondary side water level is greater than 74% or, if WR channel is inoperable, narrow range (NR) secondary side water level is greater than 30%

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE**, or

b. The ~~narrow~~ wide range (WR) secondary side water level of at least two steam generators shall be greater than ~~10%~~. 74% or, if WR channel is inoperable, narrow range (NR) secondary side water level shall be greater than 30%

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The ~~narrow range~~ secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

---

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

***A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 325°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures. Delete 1

 Delete

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

Delete Delete Delete

4.4.5.2 Steam Generator Tube* Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Tables 4.4-2 ~~A, B and C~~. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

   2. Tubes in those areas where experience has indicated potential problems, and

~~* When referring to a steam generator tube, the sleeve shall be considered part of the tube if the tube has been repaired per Specification 4.4.5.4.a.12.~~

Delete

4.4.5.2 (Continued)

3. A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

Delete

c. The tubes selected as the second and third samples (if required by Tables 4.4-2 A, B and C) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2. The inspections include those portions of the tubes where imperfections were previously found.

Delete

d. Each inspection shall include a sample of those tubes expanded in the preheater section of the steam generator. The first sample size, second sample size and subsequent inspection shall follow Table 4.4-2B.

e. In addition to the 3% sample, all tubes for which the alternate plugging criteria (F*) has been previously applied shall be inspected in the tubesheet region.

f. When applying the exceptions of 4.4.5.2.a through 4.4.5.2.e, previous defects or imperfections in the area covered by the sleeve pressure boundary are not considered an area requiring reinspection.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Delete

Note: In all inspections, previously degraded tubes or sleeves must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

Delete

Steam Generator Replacement

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of ~~initial criticality~~. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Tables 4.4-2 ~~A and B~~ at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

Delete

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Tables 4.4-2 ~~A and B~~ during the shutdown subsequent to any of the following conditions:

Delete

1. Reactor-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

2. A seismic occurrence greater than the Operating Basis Earthquake, or

3. A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

4. A main steam line or feedwater line break.

Add

Amendment No.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1. Imperfection means an exception to the dimensions, finish, or contour of a tube ~~or sleeve~~ from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube ~~or sleeve~~ wall thickness, if detectable, may be considered as imperfections;

2. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube ~~or sleeve~~;

3. Degraded Tube means a tube~~, including the sleeve if the tube has been repaired,~~ containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

4. % Degradation means the percentage of the tube ~~or sleeve~~ wall thickness affected or removed by degradation;

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube ~~or sleeve~~ containing a defect is defective;

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be ~~repaired, (i.e. sleeved) or~~ removed from service and is equal to 40% of the nominal tube wall thickness. ~~Plugging will be required for all tubes that have been repaired by sleeving where degradation in the sleeve pressure boundary is detected. The portion of the tube and sleeve for which indications of wall degradation must be evaluated is that portion constituting the pressure boundary. The pressure boundary is defined as follows:~~

~~i. For a free span sleeve (both ends welded), the pressure boundary consists of that portion between the upper and lower weld centerlines.~~

~~ii. For a tubesheet area sleeve (lower end rolled), the pressure boundary consists of that portion between the upper weld centerline and the lower rolled joint centerline.~~

~~iii. For the tube, the pressure boundary consists of those portions of the tube above (and below for transition zone sleeves) the sleeve weld centerline.~~

~~The plugging limit does not apply to the area of the tubesheet region below the F* distance provided the tube is not degraded (i.e., no indications of cracking) within the F* distance.~~

Delete

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria (Continued)

7. Unserviceable describes the condition of a tube ~~or sleeve~~ if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.4.5.3c., above;

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. ~~For a tube that has been repaired by sleeving, the tube inspection should include the sleeved portion of the tube;~~

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator ~~, including the sleeve if the tube has been repaired,~~ performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to ~~initial~~ POWER OPERATION using ~~the~~ equipment and techniques expected to be used during subsequent inservice inspections.

with the replacement steam generators

~~10. F* Distance is the distance into the tubesheet from the face of the tubesheet or the top of the last hardroll, whichever is lower (further into the tubesheet), that has been conservatively chosen to be 1.6 inches.~~

~~11. Alternate Tube Plugging Criteria does not require the tube to be removed from service or repaired when the tube degradation exceeds the plugging limit so long as the degradation is in that portion of the tube from F* to the bottom of the tubesheet. This definition does not apply to tubes with degradation (i.e., indications of cracking) in the F* distance.~~

~~12. Tube Repair refers to sleeving, as described in Combustion Engineering Report CEN-630-P, Rev. 01 which is used to maintain a tube in service or return a tube to service. This includes the removal of plugs that were installed as a corrective or preventative measure. A post-weld heat treatment during installation will be performed. 100% of the weld zones will be visual test (VT) inspected upon installation.~~

Delete

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria (Continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug ~~or repair~~ all tubes exceeding the plugging limit ~~and all tubes containing through-wall cracks)~~ required by Table~~s~~ 4.4-2~~A, B and C~~.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged ~~or repaired~~ in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube ~~(including sleeves)~~ inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1. Number and extent of tubes ~~and sleeves~~ inspected,

2. Location and percent of wall-thickness penetration for each indication of an imperfection, and

3. Identification of tubes plugged ~~or repaired~~.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

~~d. The results of the inspection of F* tubes shall be reported to the Commission in a report, prior to the restart of the unit following the inspection. This report shall include:~~

~~1. Identification of F* tubes, and~~

~~2. Location and size of the degradation.~~

~~NRC approval of this report is not required prior to restart.~~

Delete

TABLE 4.4-2A Delete

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S.G. Delete	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G. Delete	C-1	None
					C-2	Plug or repair defective tubes Delete
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A Delete
	C-3	Inspect all tubes in this S.G., plug or repair defective tubes and inspect 2S tubes in each other S.G. Delete Notification to NRC pursuant to Specification 4.4.5.5.c.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G.s are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or repair defective tubes. Delete Delete Notification to NRC pursuant to Specification 4.4.5.5.c.	N/A	N/A

$S = \frac{9}{n}\%$ where n is the number of steam generators inspected during an inspection.

* Defective tubes which fall under the Alternate Tube Plugging Criteria do not have to be plugged or repaired. Delete

TABLE 4.4-2B

STEAM GENERATOR TUBE INSPECTION - TUBE EXPANDED IN PREHEATER REGION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of S of the tubes expanded in the preheater section	C-1	None	N/A	N/A
	C-2	Plug or repair defective tubes and inspect all other expanded tubes in this Steam Generator	C-1	N/A
			C-2	Plug or repair defective tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all expanded tubes in this Steam Generator, plug or repair defective tubes and inspect all expanded tubes in each other Steam Generator.	All other S.G.s are C-1	None
			One or more S.G.s C-2 but no additional S.G.s are C-3	Plug or repair defective tubes
		Notification to NRC pursuant to Specification 4.4.5.5.c.	Additional S.G. is C-3	Plug or repair defective tubes. Notification to NRC pursuant to Specification 4.4.5.5.c.

$S = \frac{9}{n}\%$ where n is the number of steam generators inspected during an inspection.

Delete

TABLE 4.4-8C

STEAM GENERATOR TUBE SLEEVE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of each type of installed sleeve per S.G. (1)	C-1	None	N/A	N/A
	C-2	Plug tubes containing defective sleeves and inspect all remaining installed sleeves in this S.G.	C-1	None
			C-2	Plug tubes containing defective sleeves
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all installed sleeves in this S.G., plug tubes containing defective sleeves and inspect 100% of the installed sleeves in each other S.G.	Each other S.G. is C-1	None
		Special Report to NRC per Specification 6.9.2	Each other S.G. is C-2	Plug tubes containing defective sleeves
			Each other S.G. is C-3	Inspect all installed sleeves in each S.G. and plug tubes containing defective sleeves Special Report to NRC per Spec. 6.9.2

(1) Each sleeve type is considered a separate popluation for determination of sample expansion.

Delete

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and (0.35)

b. Less than or equal to 100/$\bar{E}$ microCuries per gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours (0.35) during one continuous time interval or exceeding ~~the limit line shown on Figure 3.4-1~~ 60.0 microCurie per gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the reactor coolant greater than 100/$\bar{E}$ microCuries per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microcurie per gram DOSE EQUIVALENT I-131 (0.35) or greater than 100/$\bar{E}$ microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

*With T_{avg} greater than or equal to 500°F.

Delete without replacement

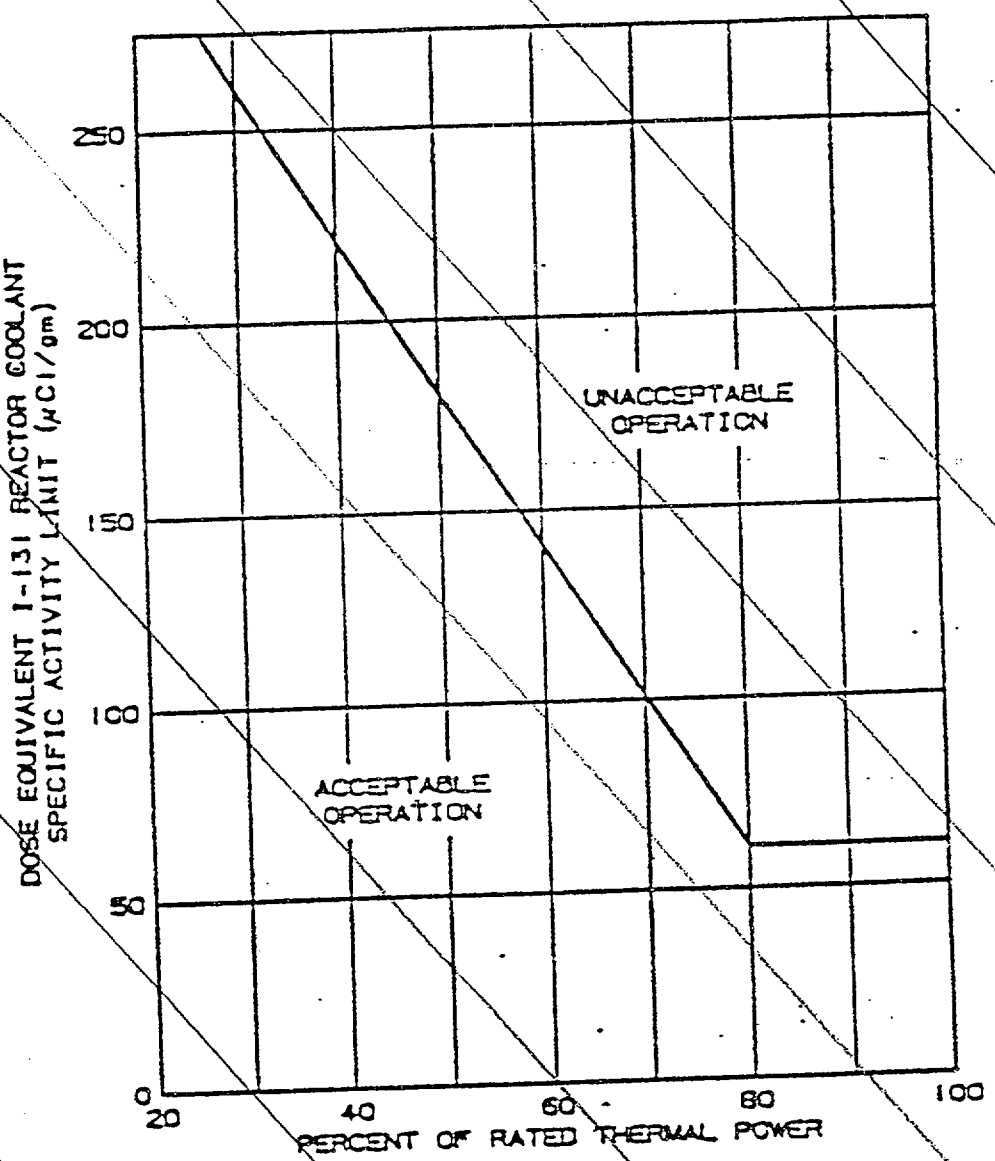


FIGURE 3.4-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 μCi/gram DOSE EQUIVALENT I-131

Add Amendment No.

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Radioactivity Determination	At least once per 72 hours.	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days.	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months**	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a. Once per 4 hours, whenever the specific activity exceeds ~~1~~ 0.35 μCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b. One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

Delete

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	~~58~~ 50
2	~~35~~ 33
3	~~17~~ 16

Delete

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

Specification 3.4.1.1

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of ~~the following requirements~~ may be suspended:

during ~~a. Specification 3.4.1.1 - During~~ the performance of startup and PHYSICS TESTS in MODE 1 or 2 provided:

a. ~~1.~~ The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. ~~2.~~ The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

~~b. Specification 3.4.1.2 - During the performance of hot rod drop time measurements in MODE 3 provided at least two reactor coolant loops as listed in Specification 3.4.1.2 are OPERABLE.~~

APPLICABILITY: During operation below the P-7 Interlock Setpoint ~~or performance of hot rod drop time measurements~~.

ACTION:

~~a.~~ With the THERMAL POWER greater than the P-7 Interlock Setpoint ~~during the performance of startup and PHYSICS TESTS~~, immediately open the reactor trip breakers.

~~b. With less than the above required reactor coolant loops OPERABLE during performance of hot rod drop time measurements, immediately open the reactor trip breakers and comply with the provisions of the ACTION statements of Specification 3.4.1.2.~~

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

~~4.10.4.3 At least the above required reactor coolant loops shall be determined OPERABLE within 4 hours prior to initiation of the hot rod drop time measurements and at least once per 4 hours during the hot rod drop time measurements by verifying correct breaker alignments and indicated power availability and by verifying secondary side narrow range water level to be greater than or equal to 41%.~~

BORATION SYSTEMS (Continued)

condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single boron injection flow path becomes inoperable.

The limitation for a maximum of one charging/safety injection pump (CSIP) to be OPERABLE and the Surveillance Requirement to verify all CSIPs except the required OPERABLE pump to be inoperable below 325°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide the required SHUTDOWN MARGIN as defined by Specification 3/4.1.1.2 after xenon decay and cooldown from 200°F to 140°F. This condition requires either 7150 ~~6650~~ gallons of 7000 ppm borated water be maintained in the boric acid storage tanks or 106,000 gallons of 2400-2600 ppm borated water be maintained in the RWST.

The gallons given above are the amounts that need to be maintained in the tank in the various circumstances. To get the specified indicated levels used for surveillance testing, each value had added to it an allowance for the unusable volume of water in the tank, allowances for other identified needs, and an allowance for possible instrument error. In addition, for human factors purposes, the percent indicated levels were then raised to either the next whole percent or the next even percent and the gallon figures rounded off. This makes the LCO values conservative to the analyzed values.

The limits on contained water volume and boron concentration of the RWST also 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 BAT minimum temperature of 65°F ensures that boron solubility is maintained for concentrations of at least the 7750 ppm limit. The RWST minimum temperature is consistent with the STS value and is based upon other considerations since solubility is not an issue at the specified concentration levels. The RWST high temperature was selected to be consistent with analytical assumptions for containment heat load.

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) the potential effects of rod misalignment on associated accident analyses are limited. 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.

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. A 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 control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The preferred sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8. If other locations must be used, a special report to NRC should be submitted within 30 days in accordance with 10CFR50.4.

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 initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR that is equal to or greater than the design DNBR value throughout each analyzed transient. The indicated T_{avg} value and the indicated pressurizer pressure value are compared to analytical limits of ~~586.4~~°F and 594.8 2185 psig, respectively, after an allowance for measurement uncertainty is included.

When RCS flow rate is measured, an additional allowance is necessary prior to comparison with the limit of Specification 3.2.5.c. Specifically for the precision calorimetric heat balance, a normal RCS flow rate error of 2.1% will be included.

Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi, raises the nominal flow measurement allowance to 2.2% for no venturi fouling. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant parameters.

Delete

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design DNBR value 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 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e.; by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

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

Insert B

In MODE 5 with reactor coolant loops not filled, a single RHR 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 RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) 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 reduction will, therefore, be within the capability of operator recognition and control.

Delete 1

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 325°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 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 380,000 lbs 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 temperatures.

Delete

TS page B 3/4 4-1

INSERT B

Surveillance Requirements for MODES 3, 4, and 5 with reactor coolant loops filled require verification of steam generator (SG) OPERABILITY. Verification of adequate level in the applicable steam generator ensures an adequate heat sink for the removal of decay heat. If the SG tubes become uncovered, the associated loop may not be capable of providing the heat sink for the removal of the decay heat. The level values include allowances for channel uncertainty and process measurement effects. The 12 hour frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

SAFETY VALVES (Continued)

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 second Reactor Trip 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.

3/4.4.3 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 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. 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 Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

In MODES 1, 2, and 3 the power-operated relief values (PORVs) provide an RCS pressure boundary, manual RCS pressure control for mitigation of accidents, and automatic RCS pressure relief to minimize challenges to the safety valves.

Providing an RCS pressure boundary and manual RCS pressure control for mitigation of a steam generator tube rupture (SGTR) are the safety-related functions of the PORVs in MODES 1, 2, and 3. The capability of the PORV to perform its function of providing an RCS pressure boundary requires that the PORV or its associated block valve is closed. The capability of the PORV to perform manual RCS pressure control for mitigation of a SGTR accident is based on manual actuation and does not require the automatic RCS pressure control function. The automatic RCS pressure control function of the PORVs is not a safety-related function in MODES 1, 2, and 3. The automatic pressure control function limits the number of challenges to the safety valves, but the safety valves perform the safety function of RCS overpressure protection. Therefore, the automatic RCS pressure control function of the PORVs does not have to be available for the PORVs to be operable.

STEAM GENERATORS (Continued)

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and ~~repaired or~~ plugged. Delete

Delete

Plugging

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. ~~Repair or plugging~~ will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. This plugging limit does not apply to imperfections located below the F* region of any given tube. The F* criterion can be applied only if the tube geometry in the region selected as the F* distance falls within the analytical limits of WCAP-12816. A sleeved tube must be plugged if degradation is detected in the sleeve pressure boundary. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission in a Special Report pursuant to Specification 4.4.5.5.c within 30 days and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 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.

3/4.4.6.2 OPERATIONAL LEAKAGE

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.

Delete

CHEMISTRY (Continued)

that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited 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.

less than 60.0 micro Curie/gram DOSE EQUIVALENT I-131

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. 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 SHEARON HARRIS site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for 0.35 limited time periods with the reactor coolant's specific activity greater than ~~1~~ 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. See Generic Letter 85-19 for additional information.

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of ~~1~~ 0.35 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 15 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 15 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a

Add Amendment No.

CONTAINMENT AIR LOCKS (Continued)

Delete

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J, as modified by approved exemptions. HNP has an approved exemption to Appendix J Option A, paragraph III.D.2 of 10 CFR 50 in that the Overall air lock leakage test is required to be performed if maintenance has been performed that could affect the air lock sealing capability prior to establishing CONTAINMENT INTEGRITY. This is in contrast to the Appendix J requirement if air locks are opened during periods when containment integrity is not required by the plant's Technical Specifications shall be tested at the end of such periods.

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 -2 psig, and (2) the containment peak pressure does not exceed the design pressure of 45 psig.

The maximum peak pressure expected to be obtained from a postulated ~~main steam line break event~~ LOCA is ~~41.2~~ 41.8 psig using a value of ~~1.9~~ 1.6 psig for initial positive containment pressure. ~~However, since the instrument tolerance for containment pressure is 1.32 psig and the high-one setpoint is 3.0 psig, the pressure limit was reduced from the high one setpoint by slightly more than the tolerance and was set at 1.6 psig. This value will prevent spurious safety injection signals caused by instrument drift during normal operation.~~ The -1" wg was chosen to be consistent with the initial assumptions of the accident analyses.

Delete

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 safety analysis for a LOCA or steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

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 maximum pressure of a postulated ~~main steam line break~~ LOCA 41.8 psig ~~accident (41.2 psig)~~. A visual inspection in conjunction with the Containment Leakage Rate Testing Program is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment preentry purge makeup and exhaust isolation valves are required to be sealed closed during plant operations in MODES 1, 2, 3 and 4 since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during these MODES ensures that excessive quantities of radioactive materials will not be released via the Pre-entry Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the Normal Containment Purge System is restricted to the 8-inch purge makeup and exhaust isolation valves since, unlike the 42-inch valves, the 8-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline of 10 CFR Part 100 would not be exceeded in the event of an accident during normal containment PURGING operation. The total time the Normal Containment Purge System isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, may be used to justify the opening of these isolation valves during MODES 1, 2, 3, and 4.

Leakage integrity tests with a maximum allowable leakage rate for containment purge makeup and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before

Delete

AUXILIARY FEEDWATER SYSTEM

maximum allowable minimum

operation. The AFW System provides decay heat removal immediately following a station blackout event, and is required to mitigate the Loss of Normal Feedwater and Feedwater Line break accidents analyzed in FSAR Chapter 15. The pump performance requirements are based upon a [4%] degradation of the vendor certified performance curves. Pump operation at this level has been demonstrated by calculation to deliver sufficient AFW flow to satisfy the accident analysis acceptance criteria.

delete Delete pump

With regard to the periodic AFW valve position verification of Surveillance Requirement 4.7.1.2.1 Sub-paragraph a.3, this requirement does not include in its scope the AFW flow control valves inline from the AFW motor-driven pump discharge header to each steam generator when they are equipped with an auto-open feature. The auto-open logic feature is designed to automatically open these valves upon receipt of an Engineered Safety Features System AFW start signal. As a consequence, valves with an auto-open feature do not have a "correct position" which must be verified. The valves may be in any position, in any MODE of operation thereby allowing full use of the AFW System for activities such as to adjust steam generator water levels prior to and during plant start-up, as an alternate feedwater system during hot standby, for cooldown operations, and to establish and maintain wet layup conditions in the steam generators.

3/4.7.1.3 CONDENSATE STORAGE TANK

6

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for [12] 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, and the value has also been adjusted in a manner similar to that for the RWST and BAT, as discussed on page B 3/4 1-3.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm reactor-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety 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 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 are consistent with the assumptions used in the safety analyses.

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 a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.

INSERT E here (a generic maximum)

Delete

TS page B 3/4 7-2

INSERT E

The Shearon Harris specific RT_{NDT} is limited to a maximum value of 10°F.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 45.0 psig and a peak air temperature of 380°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly normally containing 264 fuel rods clad with Zircaloy-4 except that limited substitution of fuel rods by filler rods consisting of Zircaloy-4, stainless steel, or by vacancies may be made in fuel assemblies if justified by a cycle-specific evaluation. Should more than a total of 30 fuel rods or more than 10 fuel rods in any one assembly be replaced per refueling a Special Report describing the number of rods replaced will be submitted to the Commission, pursuant to Specification 6.9.2, within 30 days after cycle startup. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 5.0 weight percent U-235. Fuel with enrichments greater than 4.20 weight percent U-235 shall contain sufficient integral burnable absorbers such that the requirement of Specification 5.6.1.a.2 is met.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 52 shutdown and control rod assemblies. The shutdown and rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium, or 95% hafnium with the remainder zirconium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

Delete 5.4.2 The total water and steam volume of the Reactor Coolant System is approximately ~~9410 ± 100~~ 10,300 cubic feet at a nominal T_{avg} of ~~580.8~~ 588.8°F.

PROCEDURES AND PROGRAMS (Continued)

Delete

k. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54 (o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program shall be in conformance with the NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, with the following exception noted:

1) The above Containment Leakage Rate Testing Program is only applicable to Type A testing. Type B and C testing shall continue to be conducted in accordance with the original commitment to 10 CFR 50 Appendix J, Option A.

41.8

The calculated peak containment internal pressure related to the design basis loss-of-coolant accident is ~~38.4~~ psig. The calculated peak containment internal pressure related to the design basis main steam line break is ~~41.2~~ psig. P_a will ~~conservatively~~ be assumed to be ~~41.2~~ psig for the purpose of containment testing in accordance with this Technical Specification.

41.3

41.8

The maximum allowable containment leakage rate, L_a at P_a, shall be 0.1 % of containment air weight per day.

The containment overall leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the combined Type B and Type C tests, and $\leq$ 0.75 L_a for Type A tests.

The provisions of Surveillance Requirement 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. However, test frequencies specified in this Program may be extended consistent with the guidance provided in Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J," as endorsed by Regulatory Guide 1.163. Specifically, NEI 94-01 has this provision for test frequency extension:

1) Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for recommended Type A testing may be extended by up to 15 months. This option should be used only in cases where refueling schedules have been changed to accommodate other factors.

The provisions of Surveillance Requirement 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

Delete

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

Delete

d. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

e. ~~Steam Line Break Methodology as defined by:~~ delete

EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," approved version as specified in the COLR.

~~ANF-84-093(P)(A), "Steam Line Break Methodology for PWRs," approved version as specified in the COLR.~~ delete

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

Delete

f. ~~SEM/PWR Large Break LOCA Evaluation Model as defined by:~~ delete

EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," approved version as specified in the COLR.

XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," approved version as specified in the COLR.

ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," approved version as specified in the COLR.

XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.

XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

(Methodology~~s~~ies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

g. XN-NF-78-44(~~P~~ NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," approved version as specified in the COLR.

(Methodology for Specification 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.2 - Heat Flux Hot Channel Factor).

Delete

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

Delete

h. ANF-88-054(P)(A), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," approved version as specified in the COLR.

(Methodology for Specification 3.2.1 - Axial Flux Difference, and 3.2.2 - Heat Flux Hot Channel Factor).

~~i. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).~~

~~(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 AND 5, 3.2.2 - Heat Flux Hot Channel Factor and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).~~

~~j. WCAP-10266-P-A, Rev. 2, "The 1981 Version of the WESTINGHOUSE ECCS EVALUATION MODEL USING THE BASH CODE", March 1987 (W Proprietary).~~

~~(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).~~

~~k. WCAP-11837-P-A, "EXTENSION OF METHODOLOGY FOR CALCULATING TRANSITION CORE DNBR PENALTIES", January 1990 (W Proprietary).~~

~~(Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).~~

i. ~~l.~~ EMF-92-081(P)(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," approved version as specified in the COLR.

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

j. ~~m.~~ EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.

(Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

K ~~n.~~ XN-NF-82-49(P)(A), "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model," approved version as specified in the COLR.

(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

Delete

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

l. ~~o.~~ EMF-96-029(P)(A), "Reactor Analysis Systems for PWRs," approved version as specified in the COLR.

(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10CFR50.4 within the time period specified for each report.

6.10 DELETED

(PAGE 6-25 DELETED)

Delete

Enclosure 5 to SERIAL: HNP-00-142

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
TECHNICAL SPECIFICATION CHANGE REQUEST
STEAM GENERATOR REPLACEMENT

PROPOSED TECHNICAL SPECIFICATION PAGE CHANGES

RETYPED PAGES

SECTION PAGE

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

Startup and Power Operation 3/4 4-1

Hot Standby 3/4 4-2

Hot Shutdown 3/4 4-4

Cold Shutdown - Loops Filled 3/4 4-6

Cold Shutdown - Loops Not Filled 3/4 4-7

3/4.4.2 SAFETY VALVES

Shutdown 3/4 4-8

Operating 3/4 4-9

3/4.4.3 PRESSURIZER 3/4 4-10

3/4.4.4 RELIEF VALVES 3/4 4-11

3/4.4.5 STEAM GENERATORS 3/4 4-13

TABLE 4.4-1 MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION 3/4 4-18

TABLE 4.4-2 STEAM GENERATOR TUBE INSPECTION 3/4 4-19

TABLE 4.4-2B (DELETED) 3/4 4-20

TABLE 4.4-2C (DELTED) 3/4 4-20a

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

Leakage Detection Systems 3/4 4-21

Operational Leakage 3/4 4-23

TABLE 3.4-1 REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES . . . 3/4 4-25

3/4.4.7 CHEMISTRY 3/4 4-26

TABLE 3.4-2 REACTOR COOLANT SYSTEM CHEMISTRY LIMITS 3/4 4-27

TABLE 4.4-3 REACTOR COOLANT SYSTEM CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS 3/4 4-28

3/4.4.8 SPECIFIC ACTIVITY 3/4 4-29

FIGURE 3.4-1 (DELETED) 3/4 4-30

TABLE 4.4-4 REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM 3/4 4-31

INDEX

3.0/4.0 BASES

SECTION	PAGE
3/4.0 APPLICABILITY	B 3/4 0-1
3/4.1 REACTIVITY CONTROL SYSTEMS	
3/4.1.1 BORATION CONTROL	B 3/4 1-1
3/4.1.2 BORATION SYSTEMS	B 3/4 1-2
3/4.1.3 MOVABLE CONTROL ASSEMBLIES	B 3/4 1-3
3/4.2 POWER DISTRIBUTION LIMITS	B 3/4 2-1
3/4.2.1 AXIAL FLUX DIFFERENCE	B 3/4 2-1
3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, AND RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR	B 3/4 2-2
FIGURE B 3/4.2-1 (DELETED)	B 3/4 2-3
3/4.2.4 QUADRANT POWER TILT RATIO	B 3/4 2-6 |
3/4.2.5 DNB PARAMETERS	B 3/4 2-6
3/4.3 INSTRUMENTATION	
3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION	B 3/4 3-1
3/4.3.3 MONITORING INSTRUMENTATION	B 3/4 3-3
3/4.3.4 (DELETED)	B 3/4 3-6
3/4.4 REACTOR COOLANT SYSTEM	
3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION	B 3/4 4-1
3/4.4.2 SAFETY VALVES	B 3/4 4-1
3/4.4.3 PRESSURIZER	B 3/4 4-2
3/4.4.4 RELIEF VALVES	B 3/4 4-2
3/4.4.5 STEAM GENERATORS	B 3/4 4-2b |
3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE	B 3/4 4-3
3/4.4.7 CHEMISTRY	B 3/4 4-4
3/4.4.8 SPECIFIC ACTIVITY	B 3/4 4-5
3/4.4.9 PRESSURE/TEMPERATURE LIMITS	B 3/4 4-6

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, Volume 3 No. 1-4, 1979 (or equivalently, Federal Guidance Report No. 11 "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA 520/1-88-020, September 1988).

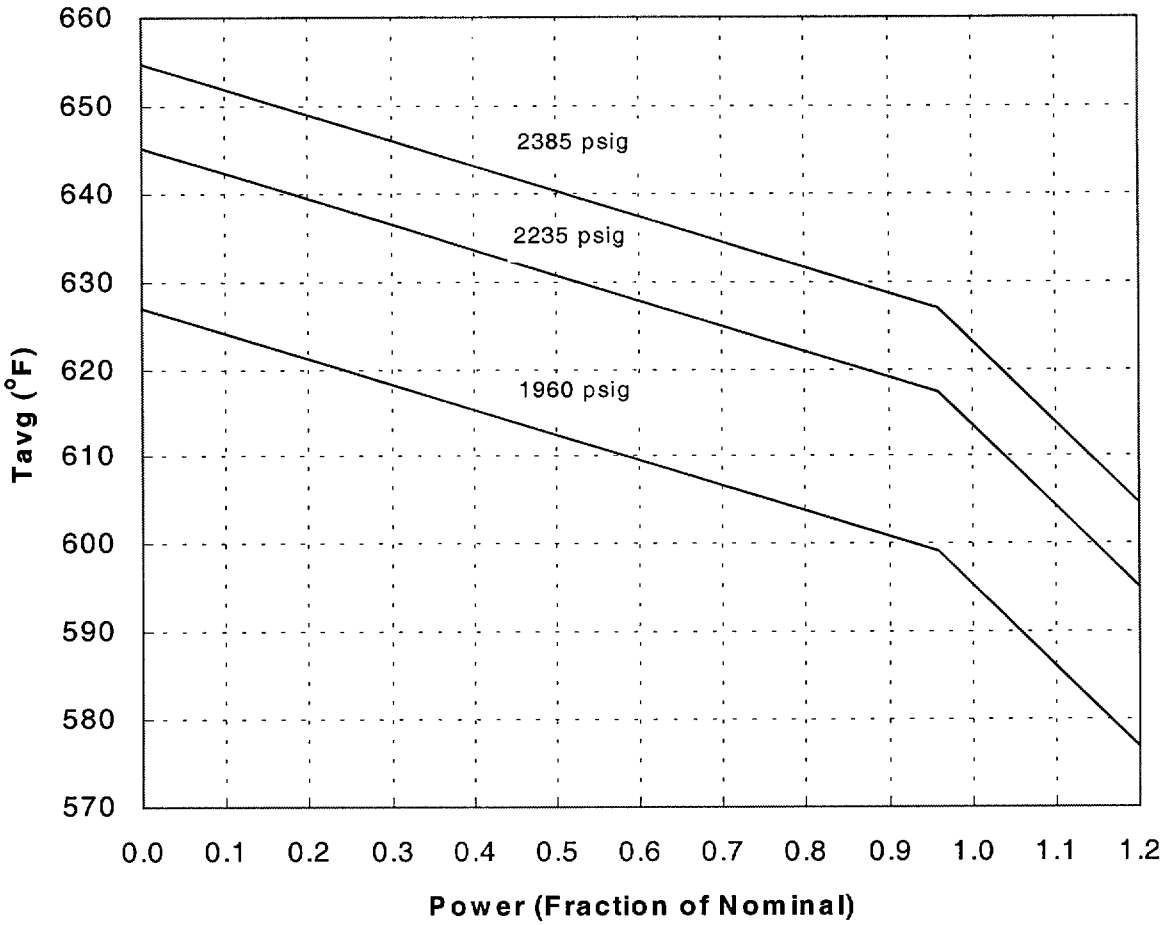


FIGURE 2.1-1
REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION
WITH MEASURED RCS FLOW > [293,540 GPM X (1.0 + C_1)]

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤ 109% of RTP**	≤ 111.1% of RTP**
b. Low Setpoint	8.3	4.56	0	≤ 25% of RTP**	≤ 27.1% of RTP**
3. Power Range, Neutron Flux, High Positive Rate	2.5	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	2.5	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% of RTP**	≤ 30.9% of RTP**
6. Source Range, Neutron Flux	17.0	10.01	0	≤ 10^5 cps	≤ 1.4 x 10^5 cps
7. Overtemperature ΔT	9.0	7.31	Note 5	See Note 1	See Note 2
8. Overpower ΔT	4.0	2.32	1.3	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	1.52	1.5	≥ 1960 psig	≥ 1948 psig
10. Pressurizer Pressure-High	7.5	1.52	1.5	≤ 2385 psig	≤ 2397 psig
11. Pressurizer Water Level-High	8.0	3.42	1.75	≤ 92% of instrument span	≤ 93.5% of instrument span

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	4.58	1.98	0.6	≥ 90.5% of loop full indicated flow	≥ 89.5% of loop full indicated flow
13. Steam Generator Water Level Low-Low	25.0	16.85	2.0	≥ 25.0% of narrow range instrument span	≥ 23.5% of narrow range instrument span
14. Steam Generator Water Level - Low Coincident With	8.9	5.35	2.0	≥ 25.0% of narrow range instrument span	≥ 23.5% of narrow range instrument span
Steam/Feedwater Flow Mismatch	20.0	3.01	Note 6	≤ 40% of full steam flow at RTP**	≤ 43.1% of full steam flow at RTP**
15. Undervoltage - Reactor Coolant Pumps	14.0	1.3	0.0	≥ 5148 volts	≥ 4920 volts
16. Underfrequency - Reactor Coolant Pumps	5.0	3.0	0.0	≥ 57.5 Hz	≥ 57.3 Hz
17. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥ 1000 psig	≥ 950 psig
b. Turbine Throttle Valve Closure	N.A.	N.A.	N.A.	≥ 1% open	≥ 1% open
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left[\frac{1}{1 + \tau_3 S}\right] \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left[\frac{1}{1 + \tau_6 S}\right] - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 0 s, τ_2 = 0 s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constants utilized in the lag compensator for ΔT, τ_3 = 4 s;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.185;
K_2	=	0.0224/°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 22 s, τ_5 = 4 s;

NOTE 1: (Continued)

T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;
T'	=	Reference T_{avg} at RATED THERMAL POWER ($\leq$588.8°F);
K_3	=	0.0012/psig;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between -21.6% and +12.0%, f_1 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -21.6%, the ΔT Trip Setpoint shall be automatically reduced by 1.75% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 12.0%, the ΔT Trip Setpoint shall be automatically reduced by 1.50% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input; 2.0% of ΔT span for T_{avg} input; 0.4% of ΔT span for pressurizer pressure input; and 0.7% of ΔT span for ΔI input.

TABLE NOTATIONS

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \frac{(1)}{(1 + \tau_3 S)} \leq \Delta T_o \left\{ K_4 - K_5 \frac{(\tau_7 S)}{(1 + \tau_7 S)} \frac{(1)}{(1 + \tau_6 S)} T - K_6 \left[T \frac{(1)}{(1 + \tau_6 S)} - T'' \right] - f_2(\Delta I) \right\}$$

Where:

ΔT	=	As defined in Note 1.
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	As defined in Note 1.
τ_1, τ_2	=	As defined in Note 1.
$\frac{1}{1 + \tau_3 S}$	=	As defined in Note 1.
τ_3	=	As defined in Note 1.
ΔT_o	=	As defined in Note 1.
K_4	=	1.12.
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature.
$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation.
τ_7	=	Time constants utilized in the rate-lag compensator for T_{avg}, τ_7 = 13 s.
$\frac{1}{1 + \tau_6 S}$	=	As defined in Note 1.
τ_6	=	As defined in Note 1.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: (Continued)

K_6	=	0.002/°F for T > T" and K_6 = 0 for T ≤ T",
T	=	As defined in Note 1,
T"	=	Reference T_{avg} at RATED THERMAL POWER (≤588.8°F),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input and 0.2% of ΔT span for T_{avg} input.

NOTE 5: The sensor error is: 1.3% of ΔT span for $\Delta T/T_{avg}$ temperature measurements; and 1.0% of ΔT span for pressurizer pressure measurements.

NOTE 6: The sensor error (in % span of Steam Flow) is: 1.1% for steam flow; 1.8% for feedwater flow; and 2.4% for steam pressure.

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. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform 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.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (in this application, the HTP correlation for Siemens Fuel. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

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

Reactor Coolant Flow (Continued)

(a power level of approximately 49% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90.5% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by the setpoint value. The Steam Generator Low Water level portion of the trip is activated when the setpoint value is reached, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Buses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint 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 1.2 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 Setpoint is reached shall not exceed 0.3 second.

On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by the loss of P-7 (a power-level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. The boric acid tank with:

1. A minimum contained borated water volume of 7150 gallons which is ensured by maintaining indicated level of greater than or equal to 23%,

2. A boron concentration of between 7000 and 7750 ppm, and

3. A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1. A minimum contained borated water volume of 106,000 gallons, which is equivalent to 12% indicated level,

2. A boron concentration of between 2400 and 2600 ppm, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume, and

3. Verifying the boric acid tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 40°F.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

a. Reactor Coolant System $T_{avg} \leq 594.8°F$ after addition for instrument uncertainty, and

b. Pressurizer Pressure $\geq$ 2185 psig* after subtraction for instrument uncertainty, and

c. RCS total flow rate $\geq$ 293,540 gpm after subtraction for instrument uncertainty.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters not within its specified limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown in Specification 3.2.5 shall be verified to be within its limit at least once per 12 hours.

4.2.5.2 Verify, by precision heat balance, that RCS total flow rate is within its limit at least once per 18 months.**

* This limit is not applicable during either a THERMAL POWER Ramp in excess of ±5% RATED THERMAL POWER per minute or a THERMAL POWER step change in excess of ±10% RATED THERMAL POWER.

** Required to be performed within 24 hours after $\geq$ 95% RATED THERMAL POWER.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-1	3.64	0.71	1.5	≤ 3.0 psig	≤ 3.6 psig
d. Pressurizer Pressure--Low	18.75	10.47	1.5	≥ 1850 psig	≥ 1838 psig
e. Steam Line Pressure--Low	4.52	0.71	2.0	≥ 601 psig	≥ 581.5 psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-3	3.64	0.71	1.5	≤ 10.0 psig	≤ 11.0 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation (Continued)					
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
4) Containment Radioactivity					
a) Area Monitors (both preentry and normal purges)	See Table 3.3-6, Item 1.a, for trip setpoint.				
b) Airborne Gaseous Radioactivity					
(1) RCS Leak Detection (normal purge)	See Table 3.3-6, Item 1.b.1, for trip setpoint.				
(2) Preentry Purge Detector	See Table 3.3-6, Item 1.b.2, for trip setpoint.				
c) Airborne Particulate Radioactivity					
(1) RCS Leak Detection (normal purge)	See Table 3.3-6, Item 1.c.1, for trip setpoint.				
(2) Preentry Purge Detector	See Table 3.3-6, Item 1.c.2, for trip setpoint.				
5) Manual Phase "A" Isolation	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. Main Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-2	3.64	0.71	1.5	≤ 3.0 psig	≤ 3.6 psig
d. Steam Line Pressure--Low	See Item 1.e. above for Steam Line Pressure--Low Trip Setpoints and Allowable Values.				
e. Negative Steam Line Pressure Rate--High	2.3	0.5	0	≤ 100 psi#	≤ 119.5 psi#**
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation (Continued)					
b. Steam Generator Water Level--High-High (P-14)	22.0	9.63	2.0	≤ 78.0% of narrow range instrument span.	≤ 79.5% of narrow range instrument span.
c. Safety Injection	See Item 1. above for Safety Injection Trip Setpoints and Allowable Values.				
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level--Low-Low	25.0	16.85	2.0	≥ 25.0% of narrow range instrument span.	≥ 23.5% of narrow range instrument span.
d. Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump	See Item 9. below for all Loss-of-Offsite Trip Setpoint and Allowable Values.				
f. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)					
g. Steam Line Differential Pressure--High	5.0	0.87	3.0	≤ 100 psi	≤ 127.4 psi
Coincident With Main Steam Line Isolation (Causes AFW Isolation)	See Item 4. above for Main Steam Line Isolation Trip Setpoints and Allowable Values.				
7. Safety Injection Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level--Low-Low	N.A.	N.A.	N.A.	≥ 23.4%	≥ 20.4%
Coincident With Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
8. Containment Spray Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST--Low-Low	See Item 7.b. above for all RWST--Low-Low Trip Setpoints and Allowable Values.				
Coincident With Containment Spray	See Item 2. above for all Containment Spray Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Loss-of-Offsite Power					
a. 6.9 kV Emergency Bus Undervoltage--Primary	N.A.	N.A.	N.A.	≥ 4830 volts with a ≤ 1.0 second time delay.	≥ 4692 volts with a time delay ≤ 1.5 seconds
b. 6.9 kV Emergency Bus Undervoltage--Secondary	N.A.	N.A.	N.A.	≥ 6420 volts with a ≤ 16 second time delay (with Safety Injection).	≥ 6392 volts with a time delay ≤ 18 seconds (with Safety Injection).
				≥ 6420 volts with a ≤ 54.0 second time delay (without Safety Injection).	≥ 6392 volts with a ≤ 60 second time delay (without Safety Injection).
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≥ 2000 psig	≥ 1988 psig
Not P-11	N.A.	N.A.	N.A.	≤ 2000 psig	≤ 2012 psig
b. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	≥ 553°F	≥ 549.3°F

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with two reactor coolant pumps in operation when the Reactor Trip System breakers are closed or with one reactor coolant pump in operation when the Reactor Trip System breakers are open:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,

APPLICABILITY: MODE 3.

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.

c. With no reactor coolant loop in operation, immediately open the Reactor Trip System breakers, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

*All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying narrow range secondary side water level to be greater than or equal to 30% at least once per 12 hours.

4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying wide range (WR) secondary side water level is greater than 74% or, if WR channel is inoperable, narrow range (NR) secondary side water level is greater than 30% at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE**, or

b. The wide range (WR) secondary side water level of at least two steam generators shall be greater than 74% or, if WR channel is inoperable, narrow range (NR) secondary side water level shall be greater than 30%.

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

---

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

***A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 325°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

   2. Tubes in those areas where experience has indicated potential problems, and

4.4.5.2 (Continued)

3. A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of Steam Generator Replacement. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Reactor-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

   2. A seismic occurrence greater than the Operating Basis Earthquake, or

   3. A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   4. A main steam line or feedwater line break.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1. Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

2. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube;

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.4.5.3c., above;

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to POWER OPERATION with the replacement of steam generators using equipment and techniques expected to be used during subsequent inservice inspections.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria (Continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

   1. Number and extent of tubes inspected,

   2. Location and percent of wall-thickness penetration for each indication of an imperfection, and

   3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to Specification 4.4.5.5.c.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G.s are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to Specification 4.4.5.5.c.	N/A	N/A

$S = \frac{9}{n}\%$ where n is the number of steam generators inspected during an inspection.

TABLE 4.4-2B Deleted

TABLE 4.4-2C Deleted

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 0.35 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/$\bar{E}$ microCuries per gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 0.35 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding 60.0 microCurie per gram DOES EQUIVALENT I-131, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the reactor coolant greater than 100/$\bar{E}$ microCuries per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 0.35 microcurie per gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

*With T_{avg} greater than or equal to 500°F.

FIGURE 3.4-1 Deleted

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Radioactivity Determination	At least once per 72 hours.	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days.	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months**	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a. Once per 4 hours, whenever the specific activity exceeds 0.35 μCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b. One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	50
2	33
3	16

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of startup and PHYSICS TESTS in MODE 1 or 2 provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

BORATION SYSTEMS (Continued)

condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single boron injection flow path becomes inoperable.

The limitation for a maximum of one charging/safety injection pump (CSIP) to be OPERABLE and the Surveillance Requirement to verify all CSIPs except the required OPERABLE pump to be inoperable below 325°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide the required SHUTDOWN MARGIN as defined by Specification 3/4.1.1.2 after xenon decay and cooldown from 200°F to 140°F. This condition requires either 7150 gallons of 7000 ppm borated water be maintained in the boric acid storage tanks or 106,000 gallons of 2400-2600 ppm borated water be maintained in the RWST.

The gallons given above are the amounts that need to be maintained in the tank in the various circumstances. To get the specified indicated levels used for surveillance testing, each value had added to it an allowance for the unusable volume of water in the tank, allowances for other identified needs, and an allowance for possible instrument error. In addition, for human factors purposes, the percent indicated levels were then raised to either the next whole percent or the next even percent and the gallon figures rounded off. This makes the LCO values conservative to the analyzed values.

The limits on contained water volume and boron concentration of the RWST also 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 BAT minimum temperature of 65°F ensures that boron solubility is maintained for concentrations of at least the 7750 ppm limit. The RWST minimum temperature is consistent with the STS value and is based upon other considerations since solubility is not an issue at the specified concentration levels. The RWST high temperature was selected to be consistent with analytical assumptions for containment heat load.

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) the potential effects of rod misalignment on associated accident analyses are limited. 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.

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. A 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 control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The preferred sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8. If other locations must be used, a special report to NRC should be submitted within 30 days in accordance with 10CFR50.4.

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 initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR that is equal to or greater than the design DNBR value throughout each analyzed transient. The indicated T_{avg} value and the indicated pressurizer pressure value are compared to analytical limits of 594.8°F and 2185 psig, respectively, after an allowance for measurement uncertainty is included.

When RCS flow rate is measured, an additional allowance is necessary prior to comparison with the limit of Specification 3.2.5.c. Specifically for the precision calorimetric heat balance, a normal RCS flow rate error of 2.1% will be included.

Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi, raises the nominal flow measurement allowance to 2.2% for no venturi fouling. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant parameters.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design DNBR value 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 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

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

Surveillance Requirements for MODES 3, 4, and 5 with reactor coolant loops filled require verification of steam generator (SG) OPERABILITY. Verification of adequate level in the applicable steam generator ensures an adequate heat sink for the removal of decay heat. If the SG tubes become uncovered, the associated loop may not be capable of providing the heat sink for the removal of the decay heat. The level values include allowances for channel uncertainty and process measurement effects. The 12 hour frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

In MODE 5 with reactor coolant loops not filled, a single RHR 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 RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) 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 reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 325°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 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 380,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any

SAFETY VALVES (Continued)

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 temperatures.

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 second Reactor Trip 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.

3/4.4.3 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 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. 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 Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

In MODES 1, 2, and 3 the power-operated relief values (PORVs) provide an RCS pressure boundary, manual RCS pressure control for mitigation of accidents, and automatic RCS pressure relief to minimize challenges to the safety valves.

Providing an RCS pressure boundary and manual RCS pressure control for mitigation of a steam generator tube rupture (SGTR) are the safety-related functions of the PORVs in MODES 1, 2, and 3. The capability of the PORV to perform its function of providing an RCS pressure boundary requires that the PORV or its associated block valve is closed. The capability of the PORV to perform manual RCS pressure control for mitigation of a SGTR accident is based on manual actuation and does not require the automatic RCS pressure control function. The automatic RCS pressure control function of the PORVs is not a safety-related function in MODES 1, 2, and 3. The automatic pressure control function limits the number of challenges to the safety valves, but the safety valves perform the safety function of RCS overpressure protection. Therefore, the automatic RCS pressure control function of the PORVs does not have to be available for the PORVs to be operable.

STEAM GENERATORS (Continued)

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission in a Special Report pursuant to Specification 4.4.5.5.c within 30 days and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 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.

3/4.4.6.2 OPERATIONAL LEAKAGE

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.

CHEMISTRY (Continued)

that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited 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.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. 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 SHEARON HARRIS 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 reactor coolant's specific activity greater than 0.35 microCurie/gram DOSE EQUIVALENT I-131, but less than 60.0 microCurie/gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. See Generic Letter 85-19 for additional information.

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 0.35 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 15 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 15 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a

CONTAINMENT AIR LOCKS (Continued)

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J, as modified by approved exemptions. HNP has an approved exemption to Appendix J Option A, paragraph III.D.2 of 10 CFR 50 in that the Overall air lock leakage test is required to be performed if maintenance has been performed that could affect the air lock sealing capability prior to establishing CONTAINMENT INTEGRITY. This is in contrast to the Appendix J requirement if air locks are opened during periods when containment integrity is not required by the plant's Technical Specifications shall be tested at the end of such periods.

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 -2 psig, and (2) the containment peak pressure does not exceed the design pressure of 45 psig.

The maximum peak pressure expected to be obtained from a postulated LOCA is 41.8 psig using a value of 1.6 psig for initial positive containment pressure. The -1" wg was chosen to be consistent with the initial assumptions of the accident analyses.

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 safety analysis for a LOCA or steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

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 maximum pressure of a postulated LOCA (41.8 psig). A visual inspection in conjunction with the Containment Leakage Rate Testing Program is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment preentry purge makeup and exhaust isolation valves are required to be sealed closed during plant operations in MODES 1, 2, 3 and 4 since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during these MODES ensures that excessive quantities of radioactive materials will not be released via the Pre-entry Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the Normal Containment Purge System is restricted to the 8-inch purge makeup and exhaust isolation valves since, unlike the 42-inch valves, the 8-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline of 10 CFR Part 100 would not be exceeded in the event of an accident during normal containment PURGING operation. The total time the Normal Containment Purge System isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, may be used to justify the opening of these isolation valves during MODES 1, 2, 3, and 4.

Leakage integrity tests with a maximum allowable leakage rate for containment purge makeup and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before

AUXILIARY FEEDWATER SYSTEM

operation. The AFW System provides decay heat removal immediately following a station blackout event, and is required to mitigate the Loss of Normal Feedwater and Feedwater Line break accidents analyzed in FSAR Chapter 15. The minimum pump performance requirements are based upon a maximum allowable degradation of the pump performance curves. Pump operation at this level has been demonstrated by calculation to deliver sufficient AFW flow to satisfy the accident analysis acceptance criteria.

With regard to the periodic AFW valve position verification of Surveillance Requirement 4.7.1.2.1 Sub-paragraph a.3, this requirement does not include in its scope the AFW flow control valves inline from the AFW motor-driven pump discharge header to each steam generator when they are equipped with an auto-open feature. The auto-open logic feature is designed to automatically open these valves upon receipt of an Engineered Safety Features System AFW start signal. As a consequence, valves with an auto-open feature do not have a "correct position" which must be verified. The valves may be in any position, in any MODE of operation thereby allowing full use of the AFW System for activities such as to adjust steam generator water levels prior to and during plant start-up, as an alternate feedwater system during hot standby, for cooldown operations, and to establish and maintain wet layup conditions in the steam generators.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 6 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, and the value has also been adjusted in a manner similar to that for the RWST and BAT, as discussed on page B 3/4 1-3.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm reactor-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety 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 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 are consistent with the assumptions used in the safety analyses.

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 a steam generator RT_{NDT} of 60°F (a generic maximum) and are sufficient to prevent brittle fracture. The Shearon Harris specific RT_{NDT} is limited to a maximum value of 10°F.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 45.0 psig and a peak air temperature of 380°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly normally containing 264 fuel rods clad with Zircaloy-4 except that limited substitution of fuel rods by filler rods consisting of Zircaloy-4, stainless steel, or by vacancies may be made in fuel assemblies if justified by a cycle-specific evaluation. Should more than a total of 30 fuel rods or more than 10 fuel rods in any one assembly be replaced per refueling a Special Report describing the number of rods replaced will be submitted to the Commission, pursuant to Specification 6.9.2, within 30 days after cycle startup. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 5.0 weight percent U-235. Fuel with enrichments greater than 4.20 weight percent U-235 shall contain sufficient integral burnable absorbers such that the requirement of Specification 5.6.1.a.2 is met.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 52 shutdown and control rod assemblies. The shutdown and rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium, or 95% hafnium with the remainder zirconium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is approximately 10,300 cubic feet at a nominal T_{avg} of 588.8°F.

PROCEDURES AND PROGRAMS (Continued)

k. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54 (o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program shall be in conformance with the NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, with the following exception noted:

1) The above Containment Leakage Rate Testing Program is only applicable to Type A testing. Type B and C testing shall continue to be conducted in accordance with the original commitment to 10 CFR 50 Appendix J, Option A.

The calculated peak containment internal pressure related to the design basis loss-of-coolant accident is 41.8 psig. The calculated peak containment internal pressure related to the design basis main steam line break is 41.3 psig. P_a will be assumed to be 41.8 psig for the purpose of containment testing in accordance with this Technical Specification.

The maximum allowable containment leakage rate, L_a at P_a, shall be 0.1 % of containment air weight per day.

The containment overall leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the combined Type B and Type C tests, and $\leq$ 0.75 L_a for Type A tests.

The provisions of Surveillance Requirement 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. However, test frequencies specified in this Program may be extended consistent with the guidance provided in Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J," as endorsed by Regulatory Guide 1.163. Specifically, NEI 94-01 has this provision for test frequency extension:

1) Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for recommended Type A testing may be extended by up to 15 months. This option should be used only in cases where refueling schedules have been changed to accommodate other factors.

The provisions of Surveillance Requirement 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

d. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

e. EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," approved version as specified in the COLR.

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

f. EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," approved version as specified in the COLR.

XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," approved version as specified in the COLR.

ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," approved version as specified in the COLR.

XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.

XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

(Methodologies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

g. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," approved version as specified in the COLR.

(Methodology for Specification 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.2 - Heat Flux Hot Channel Factor).

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

h. ANF-88-054(P)(A), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," approved version as specified in the COLR.

(Methodology for Specification 3.2.1 - Axial Flux Difference, and 3.2.2 - Heat Flux Hot Channel Factor).

i. EMF-92-081(P)(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," approved version as specified in the COLR.

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

j. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.

(Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

k. XN-NF-82-49(P)(A), "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model," approved version as specified in the COLR.

(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

1. EMF-96-029(P)(A), "Reactor Analysis Systems for PWRs," approved version as specified in the COLR.

   (Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10CFR50.4 within the time period specified for each report.

6.10 DELETED

(PAGE 6-25 DELETED)