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

BALANCE OF PLANT (BOP) LICENSING REPORT

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ACRONYMS

ACP	Auxiliary Control Panel
AFS	Auxiliary Feedwater System
AFW	Auxiliary Feedwater
AISC	American Institute of Steel Construction
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transients without Scram
AV	Allowable Values
AWS	American Welding Society
BAT	Boric Acid Tank
BEP	Best Efficient Point
BHP	Brake Horsepower
BOP	Balance-of-Plant
BRS	Boron Recycle System
BTRS	Boron Thermal Regeneration System
BWR	Boiling Water Reactor
CCW	Component Cooling Water
CCWS	Component Cooling Water System
CPIS	Containment Purge Isolation Signal
CP&L	Carolina Power & Light Company
CR	Control Room
CSAS	Containment Spray Actuation Signal
CSIP	Charging Safety Injection Pump
CSS	Containment Spray System
CST	Condensate Storage Tank
CT	Cooling Towers
CVC	Containment Ventilation and Cooling
CVCS	Chemical and Volume Control System
CWS	Circulating Water System
DBA	Design Basis Accident
DBD	Design Basis Document
DBE	Design Basis Earthquake
DECLG	Double-Ended Cold Leg Guillotine
DEHLG	Double-Ended Hot Leg Guillotine
DEPSLG	Double-ended Pump Suction Leg Guillotine
DER	Double-Ended Rupture
DHRAM	Digital High Range Air Monitors
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generators

ACRONYMS

EOF	Emergency Operations Facility
EOP	Emergency Operating Procedures
EPRI	Electric Power Research Institute
EQ	Equipment Qualification
EQDP	Environmental Qualifications Data Package
ESCWS	Essential Services Chilled Water System
ESF	Engineered Safeguards Features
ESR	Engineering Service Requests
ESS	Extraction Steam System
ESW	Emergency Service Water
ETSB	Effluent Treatment Systems Branch
EWT	Entering Water Temperature
FBS	Filter Backwash System
FCV	Flow Control Valve
FHB	Fuel Handling Building
FOA	Forced Oil Air
FPCCS	Fuel Pool Cooling and Cleanup System
FPCS	Fuel Pool Cooling System
FRV	Flow Regulator Valve
FSAR	Final Safety Analysis Report
FWIV	Feedwater Isolation Valve
FWLB	Feedwater Line Break
FWP	Feedwater Pump
FWS	Feedwater System
GDC	General Design Criteria
GWPS	Gaseous Waste Processing System
HEPA	High-Efficiency Particulate Air
HEI	Heat Exchanger Institute
HELB	High Energy Line Break
HHSI	High-Head Safety Injection
HNP	Harris Nuclear Plant
HP	High Pressure
HVAC	Heating, Ventilating, and Air Conditioning
HX	Heat Exchanger
I&C	Instrumentation and Control
IEEE	Institute of Electrical and Electronic Engineers
IOECCS	Inadvertent Operation of the Emergency Core Cooling System
ISA	Instrument Society of America
ISI	In-Service Inspection
IST	In-Service Testing

ACRONYMS

LBB	Leak-Before-Break
LCV	Level Control Valve
LHSS	Laundry and Hot Shower Treatment System
LHSI	Low-Head Safety Injection
LOCA	Loss-of-Coolant Accident
LOOP	Loss of Offsite Power
LP	Low Pressure
LPZ	Low Population Zone
LSSS	Limiting Safety System Setting
LTOPS	Low Temperature Overpressure Protection System
LWPS	Liquid Waste Processing System
MCA	Maximum Credible Accident
MCB	Main Control Board
MCC	Motor Control Center
MCES	Main Condenser Evacuation System
MFCV	Main Feedwater Control Valve
MFIV	Main Feedwater Isolation Valve
MOIV	Motor Operated Isolation Valve
MOV	Motor-Operated Valve
MS	Main Steam
MSR	Moisture Separator Reheater or Maximum Steaming Rate
MSS	Main Steam System
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSR	Moisture Separator Reheater
MSS	Main Steam System
MSSS	Main Steam Supply System
MSSV	Main Steam Safety Valve
NESCWS	Non-Essential Services Chilled Water System
NPDES	National Pollutant Discharge Elimination System
NPS	Nominal Pipe Size
NPSH	Net Positive Suction Head
NR	Narrow Range
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply Systems
NSW	Normal Service Water
OBE	Operating Basis Earthquake
ODCM	Off-Site Dose Calculation Manual
PCSR	Permanent Cavity Seal Ring
PORV	Power-Operated Relief Valve

ACRONYMS

PRA	Probabilistic Risk Assessment
PTF	Pressure, Temperature and Flow
PWR	Pressurized Water Reactor
RAB	Reactor Auxiliary Building
RB	Reactor Building
RCB	Reactor Containment Building
RCCA	Rod Cluster Control Assembly
RCL	Reactor Coolant Loop
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RHRS	Residual Heat Removal System
RPS	Reactor Protection System
RSG	Replacement Steam Generators
RTD	Resistance Temperature Detector
RVLIS	Reactor Vessel Level Indication System
RWST	Refueling Water Storage Tank
SAF	Single Active Failure
SAT	Spray Additive Tank
SBO	Station Blackout
SER	Safety Evaluation Report
SDS	Steam Dump System
SFPC	Spent Fuel Pool Cooling
SFPCCS	Spent Fuel Pool Cooling and Cleanup System
SG	Steam Generator
SGBS	Steam Generator Blowdown System
SGR	Steam Generator Replacement
SGTP	Steam Generator Tube Plugging
SGTR	Steam Generator Tube Rupture
SHNPP	Shearon Harris Nuclear Power Plant
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SIS	Safety Injection System
SRP	Standard Review Plan
SRV	Safety Relief Valve
SSCs	Systems, Structures and Components
SUT	Start-Up Transformers
SW	Service Water
SWPS	Solid Waste Processing System
SWTS	Secondary Waste Treatment System

ACRONYMS

TAD	Total Allowable Drifts
TDH	Total Developed Head
TEMA	Tubular (Exchanger) Manufacturer's Association
T/G	Turbine/Generator
TID	Total Integrated Dose, or Technical Information Document
TSC	Technical Support Center
TTV	Turbine Throttle Valve
UAT	Unit Auxiliary Transformers
UHS	Ultimate Heat Sink
VCT	Volume Control Tank
WA	Work Authorization
WBS	Work Breakdown Structure
WGI	Washington Group International
WHES	Waste Holdup and Evaporation System
WPB	Waste Processing Building
WR	Wide Range

STEAM GENERATOR REPLACEMENT/POWER UPRATE PROJECT

BALANCE OF PLANT (BOP) LICENSING REPORT

EXECUTIVE SUMMARY

The attached Balance of Plant (BOP) Licensing Report was prepared to support the Steam Generator Replacement and Power Uprate Project (SGR/Uprate) for the Harris Nuclear Plant (HNP). CP&L has conducted engineering analyses and evaluations to determine the acceptability of replacing the original HNP Model D4 steam generators with Model Delta 75 steam generators. Where possible, analyses and evaluations were also performed to support the option for an increase of the Nuclear Steam Supply System (NSSS) thermal power output from 2787.4 MWt to 2912.4 MWt.

The NSSS analyses and evaluations prepared to support SGR/Uprate are described and summarized in the NSSS Licensing Report. The SGR/Uprate program included analyses of Loss of Coolant Accident (LOCA) and non-LOCA transients, and evaluations of the thermal hydraulic, nuclear, and mechanical fuel design aspects of the NSSS and BOP Structures, Systems, and Components (SSCs).

This BOP Licensing Report provides descriptions of the analyses and evaluations conducted to support the HNP SGR/Uprate. The BOP Licensing Report discusses the design of BOP SSCs and their functional capabilities. The BOP analyses and evaluations demonstrate that applicable acceptance criteria for BOP SSCs are met.

2.0 BOP PROGRAM DESCRIPTION

2.1 BOP Program Overview

2.1.1 Introduction and Background

The Steam Generator Replacement (SGR)/Uprating Analysis and Licensing Project has been developed to support CP&L in licensing the replacement of the existing Model D4 Steam Generators (SGs) with Model Delta 75 Replacement Steam Generators (RSGs). In order to determine the impact on major plant design features, systems and safety analyses, a detailed programmatic review was conducted by CP&L and its subcontactors.

This Balance of Plant (BOP) Licensing Report describes the evaluation of BOP Structures, Systems, Components (SSCs) and functions with consideration given to the proposed Steam Generator Replacement. The BOP Licensing Report complements the Nuclear Steam Supply System (NSSS) Licensing Report as well as other information presented in this License Amendment Application.

2.1.2 Quality Assurance and Code Requirements

The analyses and evaluations, which support the Steam Generator Replacement were performed in accordance with applicable requirements of the 10CFR50 Appendix B Quality Assurance Program (QAP). These analyses and evaluations also conform to applicable industry codes and standards in accordance with the design bases, partially described in the HNP Final Safety Analysis Report (FSAR).

2.1.3 Scope of Review

The NSSS Licensing Report describes the results of the evaluations of NSSS SSCs. This includes transient and accident analyses, nuclear fuel analyses, and evaluations of systems, components and associated functions. This BOP Licensing Report describes the evaluations of the design and licensing aspects of BOP SSCs, functions and analyses not included in the NSSS Licensing Report. The BOP evaluations and analyses utilize input from the NSSS evaluations and analyses as required.

The BOP Licensing Report, like the NSSS Licensing Report, reflects technical interface information supplied by CP&L and its subcontractors. Interfaces were developed jointly among the involved organizations, and have been corroborated by CP&L.

2.2 Condensate and Feedwater Systems

The Harris Nuclear Plant (HNP) Condensate and Feedwater Systems have been evaluated to determine their performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.2.1 Introduction and Background

The Condensate and Feedwater System is composed of the following:

- dual pressure zone condenser with a common hotwell
- two 50% strings of low-pressure feedwater heaters
- a set of full flow Condensate polishers
- two condensate pumps
- two variable speed condensate booster pumps
- two main feedwater pumps
- two 50% heater drain pumps
- two strings of high pressure feedwater heaters
- three strings of Main Feedwater Isolation and Control Valves.

The current feedwater configuration directs flow to the SGs via both the main feedwater nozzle and the auxiliary feedwater nozzles.

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change condensate and feedwater systems design function to provide flow to the Steam Generators. The SGR/Uprate will change the piping configuration within the SG cubicles, flow rate, supply pressures, and temperatures of the process fluid.

The original design of the main feedwater system required that the main feedwater pumps be able to support a 100% Load Rejection without a reactor or turbine trip. This criteria required that the pumps be able to provide 96% of the full power flow at a SG pressure 80 psi higher than the 100% power value.

Configuration changes and revised process conditions described below affect the condensate and feedwater system.

<u>Configuration Changes</u>:
The installation of the RSGs and systems evaluation for the SGR/Uprate reflect the following modifications:

- Rerouting of main feedwater piping inside the bio-shield wall;
- Removal of pre-heater bypass piping, valves and associated instrumentation;
- Removal of orifice plates, pipe flanges, and associated instrumentation upstream of the main feedwater isolation valves (MFIVs) since split flow is no longer required for the new RSG design;
- Evaluation of the existing feedwater pump impellers to meet the new head and flow requirements

Revised Process Conditions:

The following illustrates the design versus expected flow rates at the Uprate conditions:

Component Parameter	Design	Uprate
Condensate Pump Flow	12,100 gpm	8891 gpm
Condensate Booster Pump Flow	12,100 gpm	8891 gpm
Heater Drain Pump Flow	5,100 gpm	4680 gpm
Feedwater Pump Flow	15,115 gpm	14800 gpm

The original design values noted above were extracted from the FSAR tables 10.4.7-1 through 4. The current operating flow rates are slightly less than the uprate values noted above.

2.2.2 Description of Analyses and Evaluations

The condensate and feedwater systems were evaluated to ensure that, following the SGR/Uprate, the systems remain capable of performing their required functions. All of the system components and piping were evaluated to ensure that the design pressures and temperatures bounded the expected conditions at SGR/Uprate.

2.2.2.1 Condensate System

The condensate pumps and condensate booster pumps flows, head, BHP, and NPSH requirements were reviewed for the SGR/Uprate.

The existing condensate polisher system is designed for normal condensate flow and maximum condensate flow under plant abnormal conditions. These process conditions bound the changes projected at the SGR/Uprate.

2.2.2.2 Feedwater System

The SGR/Uprate will require increases in flows and pressure from those required for the current steam generators at the current power level. These changes, along with the piping changes in the supply to the new SGs, required reanalysis of the system with respect to issues such as:

- Waterhammer potential due to rapid closure of the MFIV
- System pressure transients due to postulated transients such as bubble collapse (feedline snapping)

The changes in SG design, along with restoration of the T_{avg} to 588.8°F, require more FW system pressure head than is potentially available with the existing pumps. Consideration is made to optimizing the FCV operating position at Uprate conditions.

The FW Heaters were evaluated for increases in tube side flow rates and the velocities. Shell side conditions were also evaluated with respect to industry standards.

2.2.3 Acceptance Criteria

The current design bases for the condensate and feedwater systems are described in the FSAR. The function of the condensate and feedwater system is to provide feedwater at the required temperature, pressure and flow rate to the steam generators. Individual components must be able to operate at Uprate conditions within their design parameters.

Initial system design required that component designs be capable of supporting a full load (100% power) rejection without a reactor / turbine trip. The plant was never tested for this capability. This requirement was reduced to the capability to tolerate a 50% load rejection from full power or partial power. This change in design capability reduces the potential for excessive design margin in the main feedwater pumps.

2.2.4 Results

As described in Section 2.2.1, the SGR/Uprate involves configuration changes and revised process conditions needed to accommodate the RSGs and power uprate conditions. The system design pressures are adequate to support the SGR/Uprate conditions. Design temperature changes were considered in piping and component evaluations and were found to be acceptable.

The existing design of the Condensate System up to the suction of the feedwater pumps is not impacted due to the new SGR/Uprate. The Condensate and Condensate Booster Pumps have been determined to be adequate to support the increases in process flows at uprate conditions.

The condensate polisher system will remain able to perform its design functions for the SGR/Uprate conditions. The increases in flow expected at uprate conditions are bounded by the original design capabilities of the system.

The Feedwater Heaters have been evaluated for the increases in both shell and tube side flows and the respective increases in heat duty expected at the Uprate conditions. The tubeside velocities were evaluated against the criteria in the Heat Exchanger Institute (HEI) Standard for Feedwater Heaters. The velocities were within normal design allowances. The shellside nozzle velocities were also determined, and were generally within the HEI requirements. The #4 heaters shell side nozzle velocities were determined to be slightly higher than the HEI standards, by approximately 0.5% and 2.6%. These excess velocities are judged to be insignificant.

The SGR/Uprate may require a modification of the FWP impellers to accommodate the SGR/Uprate flows and pressure at full power conditions. The feedwater analysis considers head and flow requirements and margin to ensure that the pump has capacity to support minor system flow transients and the capability to support a 50% load rejection. Feedwater Control Valve position and differential pressure will be optimized in the final design to ensure system stability and response.

The main feedwater isolation valves (MFIVs) were evaluated and are adequate for the SGR/Uprate. Removal of feedwater pre-heater bypass piping and feedwater split flow piping per SGR/Uprate, will eliminate several containment isolation valves (CIVs).

2.2.5 Conclusions

As described, the revised process conditions for the condensate and feedwater systems under SGR/Uprate conditions remain within the system design limits. Configuration changes will be implemented to accommodate the RSGs and the revised process conditions resulting from SGR/Uprate.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS thermal power level of 2787.4 MWt.

2.2.6 References

1. HNP Final Safety Analysis Report 10.4.7

2.3 Steam Systems

The Harris Nuclear Plant (HNP) Steam Systems have been evaluated to determine their performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.3.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the Main Steam System (MSS) conveys steam from the three steam generators (SGs) to the Turbine/Generator (T/G) and other equipment. The Steam Dump System is provided to alleviate transients on the NSSS following large load reductions. The Extraction Steam System is designed to enhance the efficiency of the plant by providing steam to the feedwater heaters. The Auxiliary Steam System provides steam for process use in the plant. Auxiliary steam can be supplied by the main or extraction steam system, or by the plant auxiliary boiler. The MSS and Steam Dump System will function during normal operations and transient conditions. The Extraction Steam System will only function during normal operations. The Auxiliary Steam System will function during normal operations but can function during outages and shutdown via the auxiliary boiler.

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change existing steam systems design functions. The existing steam systems remain capable of satisfying regulatory commitments in accordance with the existing FSAR.

Configuration Changes:

There are no configuration changes associated with the steam systems under the SGR/Uprate conditions.

Revised Process Conditions:

The SG outlet conditions of pressure, temperature, and flow have increased for the SGR/Uprate. A comparison table describing the pressure, temperature and flow conditions before and after the SGR/Uprate is provided in Section 2.3.2.

The impact of pressure, temperature and flow conditions to the steam systems under the SGR/Uprate conditions is addressed in Section 2.3.2.

2.3.2 Description of Analyses and Evaluations

The steam systems were evaluated to ensure that, following the SGR/Uprate, the systems remain capable of performing required functions in accordance with the existing licensing bases specified in the FSAR.

Heat balance evaluations were performed to compare the existing steam systems design bases with the SGR/Uprate operating data conditions. These evaluations include hand calculations and those modeled by computer.

The original pipe stress analysis of the steam systems utilized the PIPESTRESS System 2010 computer program. The revised stress analyses utilize more recent computer programs (i.e., ADLPIPE).

Main Steam System (MSS):

The MSS extends from the secondary side nozzles of the SGs up to and including the turbine stop valves. The MSS includes the isolation valves, isolation bypass valves, steam dump valves, power operated relief valves, safety valves, relief valves and the steam line to the auxiliary feedwater pump turbine. The steam line to the auxiliary feedwater pump is discussed in FSAR Section 10.3.1.d and FSAR Section 10.4.9.

As shown below, the SG outlet conditions of pressure, temperature, and flow increase for the SGR/Uprate conditions.

Steam Generator	**Pre-SGR/U/prate Condition - 100%**	**Post-SGR/Uprate Condition**
Flow	12,118,000 lbm/Hr	12,721,840 lbm/Hr
Pressure at SG Outlet	883 psia	1011 psia
Temperature	529.7^{0}F	545.9^{0}F

The impact of these new parameters to piping and various valves are as follows:

Main Steam System (MSS) Lines:

The MSS is designed for a pressure of 1200 psia and temperature of 600°F. These values bound the MSS operating pressure and temperature at SGR/Uprate operating conditions. The MSS lines are adequately sized for the SGR/Uprate conditions and they remain in accordance with the FSAR.

Main Steam Safety Valves (MSSVs):

The criterion to size the valves is to relieve 105% of maximum calculated steam flow at an accumulation not exceeding 110% of MSS design pressure. Each MSSV shall be demonstrated operable with lift settings as specified in FSAR Table 10.3.1-1. Therefore, the MSSVs are adequately designed for the SGR/Uprate. These valves remain in accordance with the description contained in the FSAR.

Main Steam Isolation Valves (MSIVs):

The valves function to prevent the uncontrolled blowdown of more than one steam generator and to minimize RCS cooldown and containment pressure to within acceptable limits following a main steam line break. To accomplish these functions, the MSIVs have been designed to close in five (5) seconds following receipt of a closure signal against steam line break flow condition in the forward direction. The SGR/Uprate increases the main steam operating pressure and temperature. The MSIVs will continue to close within five (5) seconds.

Rapid closure of the MSIVs following postulated steam line break causes a significant differential pressure across the valves seats and a thrust load on the main steam system piping and piping supports in the area of the MSIVs. The worst case is bounded by the throat area of the SG flow restrictors, valve seat bore and no-load operating pressure. Since none of these change with the SGR/Uprate, there is no impact.

Main Steam Isolation Bypass Valves:

The function of the Main Steam Isolation Bypass Valves at no-load and low load power conditions is to warm up the main steam lines and equalize pressure across MSIVs prior to opening of the MSIVs. SGR/Uprate has no adverse affect on the main steam conditions at these no-load and low power levels. Therefore, the Main Steam Isolation Bypass Valves have been adequately designed for the SGR/Uprate and they remain in accordance with the FSAR.

Power Operated Relief Valves [PORVs]:

The PORVs provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere. Use of the PORVs in conjunction with the Auxiliary Feedwater System permit the plant to be cooled down from the pressure set point of the lowest MSSVs to the point where the Residual Heat Removal System can be placed in service. The capacity of all PORVs is sufficient to permit cooling of the plant at the existing design. The PORV design capacities also bound the decay heat removal load demand by the SGR/Uprate. Therefore, the PORVs have been adequately designed for the SGR/Uprate and they remain in accordance with the FSAR.

Steam Dump System:

The Steam Dump System creates an artificial steam load by dumping steam from ahead of the turbine stop valves to the main condenser. It does this by releasing steam to the atmosphere and the condenser, depending upon the size of the load rejection. The system consists of eight atmospheric steam dump valves, which dump steam into the atmosphere, and six condenser dump valves, which allow steam to bypass the turbine and dump into the condenser. The Steam Dump System has no safety-related function.

The Steam Dump System was initially designed to allow the plant to accept a sudden load rejection up to 100 percent external load without incurring a reactor trip or lifting Main Steam

Safety Valves. Under the SGR/Uprate, the steam dump piping to the condenser is designed to pass up to 35% of the total steam flow and the condenser is capable of receiving up to 40% of the total steam flow. The Atmospheric dump valves provide additional capacity to meet the steam dump requirement criteria for a 100% load rejection, even though Harris has reduced this requirement.

The steam dump lines are adequately sized for the SGR/Uprate and the system remains in accordance with the existing FSAR, as described above.

Extraction Steam System:

Based upon evaluation of the extraction steam flow rates and other pertinent parameters at post-SGR/Uprate and pre-SGR/Uprate conditions, the existing design, operation and structural integrity of the Extraction Steam System are not affected by the SGR/Uprate. The calculated pressure losses in the extraction steam lines to Heaters 1, 2, 3, 4, & 5 are less than the pressure loss considered in the heat balance(s) for the SGR/Uprate. The steam line velocities for the extraction lines are within normal design limits for the application, and that the lines will be reviewed with respect to the Flow Assisted Corrosion Monitoring program. Maximum operating pressures and temperatures of all extraction steam piping at SGR/Uprate conditions are less than the existing piping design pressures and temperatures.

In addition, the increase in flow rate due to the SGR/Uprate has no impact on the operation of reverse current valves. The motor operated isolation valves are considered adequate for the SGR/Uprate conditions since the capability of these valves and their operators is also bounded by the original design requirements. The SGR/Uprate has no impact on the operation of the system during start-up and low load operations since the design operating conditions remain unchanged. The design conditions for the Extraction Steam System including feedwater heaters have not changed for the SGR/Uprate condition and the impact on the safety valves for the Extraction Steam System and on the shell side of the feedwater heaters is minimal.

As addressed above, the existing extraction steam system has been adequately designed for the SGR/Uprate and this system remains in accordance with the FSAR.

Auxiliary Steam System:

The critical parameters potentially impacted by SGR/Uprate are the source pressure and temperature from the extraction/main steam systems. Based on the heat balances (SGR/Uprate) evaluation, the minimal increases in pressure and temperature have no impact on the functional capability of the Auxiliary Steam System. The design of piping and the pressure control valve PCV 0104 (5AS-P1-1) is adequate for the SGR/Uprate conditions. Therefore, the existing auxiliary steam system has been adequately designed for the SGR/Uprate and this system remains in accordance with the FSAR.

The results of the evaluation of the steam systems and their individual components to satisfy applicable design and licensing bases.

2.3.3 Acceptance Criteria

The licensing bases for the steam systems are described in the FSAR 10.3.1. The Steam Systems are designed to perform the following functions:

Deliver steam from the three steam generators (SGs) to the turbine generator at maximum guaranteed condition; remove heat generated by the NSSS in the event the turbine generator is not in service by use of the Steam Dump System, or by relieving to atmosphere through the main steam safety valves, or the power operated relief valves; provide steam for the moisture separator reheaters (MSRs); deliver steam to the auxiliary feedwater pump turbine in the Auxiliary Feedwater System (AFS); provide steam for the Auxiliary Steam Supply System; provide steam to the Turbine Gland Sealing System; isolate the steam generators from the remaining portions of the MSSS and from each other; and provide extraction steam to the feedwater heaters.

The design, operation, and functional capabilities of the steam systems described in the FSAR, and as affected by the SGR/Uprate, were evaluated against the functional requirements.

2.3.4 Results

The SGR/Uprate does not change the system and component functions. The steam systems, with consideration given the SGR/Uprate, are evaluated below against the acceptance criteria.

The SGR/Uprate will not affect the operability of the MSSVs and the MSIVs. In addition, the SGR/Uprate will not impact the MSIV closure time.

The steam systems components were evaluated to determine their compatibility with the revised process conditions. The steam systems piping, valves and pressure boundary components are adequate for the SGR/Uprate conditions since the original design capacities of these components bound the SGR/Uprate conditions.

The location of steam systems components inside the plant remains unchanged. The design and operating conditions are either consistent with or bounded by the existing design.

The steam flow rate will increase by approximately 4.2% for the SGR/Uprate conditions. The expected change in erosion/corrosion is expected to be minimal. The increase in flow rates, while small, will continue to be monitored for their affects on the piping under the existing plant FAC program.

The normal 100% power operating and transient pressure and temperature parameters have changed for the SGR/Uprate condition. The 4.2% increase in steam flow has been evaluated and the steam systems remain in accordance with the existing design bases. There is no increase in the frequency of occurrence, and loads remain within the piping stress analysis acceptance criteria.

The MSIV functions are not affected by the SGR/Uprate. The main steam system normal operating pressure and temperature will be higher for SGR/Uprate operating conditions. The increase in flow rate will not adversely affect operating characteristics and functions credited in the plant accident analyses.

Steam systems instrumentation and control system design functions and operational characteristics are not changed by the SGR/Uprate. Existing control functions and protective features will be maintained to ensure steam systems performance in accordance with the existing design and licensing bases. The changes in system pressures and temperatures due to SGR/Uprate are minimal.

The changes in system pressures, temperatures, and flows are small for the SGR/Uprate conditions. Existing motors for the motor operated valves are adequate since the changes in the Brake Horse Power (BHP) requirements for these motors are negligible and therefore, there is no impact on the existing margin based on the name plate Horse Power (HP). Hence, changes to electrical loads for equipment and/or cables are not anticipated for the steam systems as a result of SGR/Uprate.

The MSS is designed for a pressure of 1200 psia and temperature of 600°F. These values bound the MSS operating pressure and temperature at SGR/Uprate operating conditions. The MSS has been adequately designed for the SGR/Uprate conditions. The main steam drains are only required during startup and shutdown. The SGR/Uprate does not adversely affect the main steam drains.

As for the steam dump system, analysis has shown that the steam dump lines are adequately sized for the SGR/Uprate. Therefore, the existing steam dump system has been adequately designed for the SGR/Uprate.

Based upon evaluation of the extraction steam flow rates and other pertinent parameters at SGR/Uprate conditions, the existing design, operation and structural integrity of the Extraction Steam System are not affected by the SGR/Uprate.

2.3.5 Conclusions

As described, the revised steam systems process conditions under SGR/Uprate conditions are within existing design limits. There are no configuration changes associated with the SGR/Uprate. Changes to the process conditions (steam flow, pressure and temperature) affecting the steam systems performance characteristics can be accommodated by the existing plant design. The existing steam systems have been adequately designed for the SGR/Uprate.

The systems meet sizing and design criteria as required to ensure that the design basis is maintained as is stated in the FSAR or as revised by these submittals. The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.3.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.4.5 Steam Generators
   3/4.6.3 Containment Isolation Valves
   3/4.7.1.1 Plant System - Safety Valves
   3/4.7.1.5 Plant System - Main Steam Line Isolation Valves
   3/4.7.2 Steam Generator Pressure/Temperature Limitation
   3/4.7.8 Snubbers

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,"

9. RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste Containing Components of Nuclear Power Plants," (Rev. 3, 02/76)

10. RG 1.29, "Seismic Design Classification," (Rev. 3, 09/78)

11. 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants"
    Criterion 2, "Design Bases for Protection Against Natural Phenomena"
    Criterion 4, "Environmental and Dynamic Effects Design Bases"
    Criterion 34, "Residual Heat Removal"

2.4 Turbine/Generator Evaluations

The Harris Nuclear Plant (HNP) Turbine/Generator has been evaluated to determine its performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.4.1 Introduction and Background

The HNP uses a Siemens-Westinghouse Turbine/Generator (T/G).

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change T/G design functions. T/G design is, however, affected by revised process conditions, as described below.

Possible Configuration Changes:
The admission arc may be revised via a DEH software change to increase its minimum admission arc from 50 to 75% to increase turbine reliability. This change, if implemented, would have the first 3 control valves opening together, rather than in sequence.

The main generator cooling and support systems (Gland Seal System) are continuing to be evaluated for potential modifications. If warranted, modifications may be made to enhance the turbine generator performance.

Revised Process Conditions
The T/G was evaluated with regard to process changes resulting from the SGR/Uprate. The increased NSSS thermal output under the SGR/Uprate results in increased steam pressure, temperature and flow rate to the main turbine. The process conditions used for evaluations of the main turbine and components were a best estimate steam generator pressure of ~1011 psia. The heat balance utilized this source pressure and estimates the supply conditions to the turbine at ~995 psia, 545.9°F, at a mass flow rate of ~12.72 x 10^6 lb/hr (throttle flow of ~11.68 x 10^6 lb/hr). Turbine Generator electrical output is expected to be ~998,703 kW at the steam conditions and condenser back-pressures of 2.83/4.05 inHg absolute exhaust.

2.4.2 Description of Analyses and Evaluations

The T/G was evaluated to ensure that, following the SGR/Uprate, the system remains capable of performing required functions. The turbine was evaluated at a flow passage of 102% of uprate flow to allow for uncertainty regarding component dimensions, various cycle conditions and plant measurements. The following equipment/components were evaluated by the OEM using the revised process conditions as applicable:

- High Pressure Turbine
  - (a) Nozzle Blocks
  - (b) Control Stage Blades
  - (c) Rotating and Reaction Blading
  - (d) Horizontal Joints

(e) Gland Rings

- Low Pressure Turbines
  (a) Stationary Blade Flutter
  (b) Rotor System Torsional Analysis
  (c) Disc Integrity and Missile Analysis
- Turbine Auxiliaries:
  (a) Lube Oil System;
  (b) Seal Oil System;
  (c) Gland Seal System;
  (d) EH Fluid System
- Moisture Separator Reheaters;
- Main Generator;
- Generator Exciter;
- Hydrogen Cooler;
- Piping and valves; and
- Associated instrumentation and controls.

The results of the evaluation of the T/G and its individual components are presented in Section 2.4.4. Acceptance criteria, relevant to the T/G, are identified in Section 2.4.3.

2.4.3 Acceptance Criteria

The Acceptance Criteria of the evaluations is to maintain the design, operation, and functional capabilities of the T/G described in the FSAR, and as affected by the SGR/Uprate. The existing design parameters must bound the projected conditions at SGR/Uprate conditions. The results of these evaluations are described in Section 2.4.4.

2.4.4 Results

The SGR/Uprate does not change the system and component functions. As described in Section 2.4.1, the SGR/Uprate involves configuration changes and revised process conditions. The T/G, with consideration given the SGR/Uprate, is evaluated below against the acceptance criteria and associated FSAR information.

High Pressure Turbine
The Nozzle Blocks, HP Blade Rings, HP Stationary Reaction Blading, HP Nozzle Chamber, HP Outer Cylinder and horizontal joint bolting, and HP Inner Gland Rings were evaluated with respect to SGR/Uprate. These components are acceptable to operate under SGR/Uprate.

The minimum arc of admission to the HP turbine is being evaluated for suitability at revised process conditions. The HP Rotating Reaction Blading, specifically the first rotating reaction row 1R on each end of the HP turbine, are being evaluated for suitability at the revised process conditions.

Low Pressure Turbines

The evaluation indicates the LP Reaction Blades and the LP Inner Cylinder and Blade Rings are acceptable for SGR/Uprate conditions. The LP has been evaluated and found acceptable for the projected SGR/Uprate conditions. Sub-components have been evaluated to ensure all are within acceptable limits for the projected conditions.

Rotor System and Torsional Analysis

The rotor system torsional analysis was reviewed considering the SGR/Uprate. The changes in steam flow and load do not change the rotor system torsional natural frequencies. The short circuit loading for rotor analysis is based on the maximum KVA of the generator, therefore the short circuit torque is not affected by the uprated turbine. The turbine coupling and overhang shafting were also reviewed for the increased steady torque at the SGR/Uprated load. Based on this review and design analysis, the TG shaft system meets all mechanical design requirements under the SGR/Uprate conditions.

Turbine Disc Integrity and Missile Analysis

The HNP LP turbine rotors are partial integral (PI), double flow rotors. The configuration of these rotors is that the first three discs (#1-#3) on each end closest to the inlet are integral with the rotor forging. The last two discs (#4 & #5) are separate pieces, which are shrunk on to the rotor body. The integral discs eliminate the disc bore to shaft interface, which was particularly susceptible to SCC. The remaining discs have good resistance to SCC and long lives due to the low temperature environment at the exhaust end of the LP.

Following the SGR/Uprate, the full power operating temperatures of the #4 and #5 discs are expected to be slightly reduced compared to the current operating conditions. Since the stress corrosion cracking growth rate decreases with decreasing temperature, missile generation probabilities would be slightly lower than those previously analyzed. Based on this study, the SGR/Uprate does not introduce any new hazards, nor does it increase the severity or probability of Turbine/Generator Missiles.

Instrumentation and Turbine Overspeed Protection

The turbine overspeed protection, instrumentation and devices that control the DEH or provide input to reactor control and protection circuits, and instrumentation and devices that trip the turbine were reviewed. The ability of these systems is not affected by the SGR/Uprate.

Moisture Separator Reheater

The MSRs were evaluated against the revised process conditions and are bounded by the existing design temperatures and pressures. While the effectiveness is projected to be slightly lower at the uprate conditions, the change in thermal performance is considered negligible. Tube vibration due to increased cycle flow rates was evaluated at the uprate conditions. The velocities were found to be well below the critical velocity (velocity at which fluidelastic vibration is initiated).

Hydrogen Cooler
The present capacity of the hydrogen cooler is potentially marginal for the hot weather months and the SGR/Uprate may increase nominal heat load. Replacement of the existing hydrogen cooler with a larger capacity hydrogen cooler is still being evaluated.

Turbine Auxiliaries
The lube oil, seal oil, and EH Fluid systems should have no significant effect due to the SGR/Uprate. The gland steam system may be modified to address current operating problems and to preclude problems at the SGR/Uprate conditions.

2.4.5 Conclusions

The revised process conditions expected at the SGR/Uprate conditions remain within the T/G design limits.

The T/G and associated systems will continue to comply with FSAR. The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.4.6 References

1. HNP Final Safety Analysis Report

2.5 Circulating Water System

The Harris Nuclear Plant (HNP) Circulating Water System (CWS) has been evaluated to determine its performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.5.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the CWS provides the main condenser with a continuous supply of cooling water for removing the heat rejected by the main turbines. The system is designed to operate continuously under various ambient weather conditions. In addition, the system serves as the preferred heat sink for normal reactor cooldown to 350°F, but has no safety function.

The CWS consists of the following major components: Main Condenser, three Circulating Water Pumps, Cooling Tower, Make-Up Water System, Chlorination Water Treatment System and piping, valves, expansion joints, and instrumentation.

The Main Condenser receives steam exhaust from the low pressure turbines and condenses it via a heat exchange with the Circulating Water. The natural draft, counter-flow, Cooling Tower serves as a heat sink for the heat rejected by the turbine cycle via the Condenser and for the heat rejected by the equipment supplied with the cooling water from the Normal Service Water System.

Configuration Changes:

A potential replacement of the existing 90-10 Cu-Ni condenser tubing with "Sea-Cure" (Reference 9) was evaluated at the SGR/Uprate conditions.

Revised Process Conditions:

The heat transfer duty of the Condenser increases following the SGR/Uprate. The potential change in tubing material to the new sea cure condenser tubing, which has a slightly lower heat transfer coefficient, results in slightly higher condenser backpressure for the same heat load. The increase in condenser heat duty due to the SGR/Uprate results in slightly higher CWS temperatures existing from the condenser which also yields slightly higher condenser backpressures.

The impact of the revised process conditions on the CWS is addressed in Section 2.5.2.

2.5.2 Description of Analyses and Evaluations

The CWS was evaluated to assure that, following the SGR/Uprate, the system remains capable of performing required functions in accordance with the existing licensing bases specified in the FSAR.

The existing 90-10 Cu-Ni condenser tubing may be replaced with "Sea-Cure" tubing. The Sea Cure condenser tubing has a slightly lower heat transfer coefficient and a larger inside diameter because of a thinner tube wall. The larger inside diameter would result in a slight increase in CWS flow through the condenser. Even though the increased flow rate will have some countering affect on the lower heat transfer coefficient, the resultant backpressures will still be slightly higher than those that would exist for the 90-10 Cu-Ni tubes for the same heat duty in the condenser.

The heat transfer duty of the Condenser following the SGR/Uprate will increase from ~6.2×10^9 Btu/hr to ~6.5811×10^9 Btu/hr.

The existing LP and HP Condenser operating pressure ranges are 1.35 to 3.80 in-Hga (for 32 to 99°F at 85% clean) and 2.9 to 5.30 in-Hga (for 32 to 99°F at 85% clean). Following the SGR/Uprate, the condenser operating pressures at a circulating water inlet temperature of ~95°F would result in the following estimated condenser backpressures (assuming 85% cleanliness) for the respective tubing:

Tubing Material	LP Condenser Zone 1	HP Condenser Zone 2
90-10 Cu-Ni	3.59 in Hg	5.1 in Hg
Sea Cure	3.72 in Hg	5.21 in Hg

Peak summer ambient temperatures currently increase the circulating water temperature producing higher operating pressure in each condenser zone and results in reduced turbine efficiency. Following the SGR/Uprate and the potential tube replacement, the peak summer operating pressures will be even higher further reducing turbine efficiency. Load can be reduced to decrease the operating pressure.

The higher condenser backpressures during summer conditions are not an operational concern, since there is still ample margin to the turbine trip setpoint of 7.5 inches Hg-absolute. Consequently, the increased Condenser backpressures during summer conditions do not significantly increase the probability of turbine trip, and the resulting plant transients. Therefore, the effect of the SGR/Uprate and potential tube replacement is reduced turbine efficiency during the peak summer conditions.

Following the SGR/Uprate, the CWS temperatures increase marginally from the pre-SGR/Uprate condition. The circulating water temperature at the Cooling Tower inlet will increase from 120°F to ~121°F, and the estimated circulating water temperature at the Cooling Tower basin will increase from 95.0°F to 95.2°F. The design temperatures (i.e., cooling tower inlet - 130°F and cooling tower basin - 115°F) bound these values.

The current design flow rate for the CWS is 483,000 gpm (161,000 gpm per pump). Should the condenser tubing be changed, the SGR/Uprate flow rate for the CWS is expected to increase to 487,600 gpm (162,533 gpm per pump), due to the increased diameter of the condenser tubing.

The projected increase in circulating water flow would have a negligible effect on the operation of the CW pumps and associated components.

The CWS pump BHP of 3,090, under the SGR/Uprate conditions, is bounded by the motor nameplate rating of 3,500, and also within the values in current electrical load calculations. Consequently, the electrical distribution system that serves the CWS remains in accordance with the FSAR.

Although the circulating water flow rate would increase following the SGR/Uprate if the condenser tubing is replaced, the line size is not impacted since the total system head loss is compensated by the total developed head of the circulating water pumps total flow.

The condenser, CWS, and support systems have been evaluated for the increased turbine exhaust flows and heat duty. While small increases are expected in evaporation, makeup, and blowdown flow rates, the slight increases have been determined to be within the capabilities of the existing plant equipment. The condenser has been evaluated for the increased turbine exhaust flows and heat duty and has been determined to be acceptable with the new tubing material.

The results of the evaluation of the CWS and its individual components to satisfy applicable design and licensing bases in accordance with the FSAR are presented in Section 2.5.4. Acceptance criteria, relevant to the CWS, are identified in Section 2.5.3.

2.5.3 Acceptance Criteria

The licensing bases for the CWS are described in the FSAR. The CWS is designed to provide the main condenser with a continuous supply of cooling water for removing the heat rejected by the main turbines.

The design, operation, and functional capabilities of the CWS described in the FSAR, and as affected by the SGR/Uprate, were evaluated against the acceptance criteria. The results of these evaluations are described in Section 2.5.4.

2.5.4 Results

The SGR/Uprate does not change the CWS and component design functions as described in the FSAR. The SGR/Uprate involves revised process conditions. Operation using the existing Cu-Ni condenser tubes or changing to the Sea Cure tubing have been considered and found acceptable from an operational standpoint. Temperature, pressure, and flow characteristics will be maintained within original design limits after SGR/Uprate.

The CWS components were evaluated to determine their compatibility with the revised process conditions. Hardware modifications to the CWS, which are part of the SGR/Uprate, are described in Section 2.5.1. All other existing system components are adequate since the original design capacities of these components bound the SGR/Uprate conditions.

2.5.5 Conclusions

The revised process conditions for the CWS under SGR/Uprate conditions remain within the system design limits. Utilizing the existing condenser tubing or the potential change to the Sea Cure tubing as described in this LR, all system components are adequate to meet the SGR/Uprate conditions. Although the SGR/Uprate does result in increased heat duty, the existing CWS design is adequate for the system to continue to function properly in accordance with existing licensing/design basis requirements. The heat removal functions provided by the CWS will continue to be achieved under SGR/Uprate conditions. The CWS, following the SGR/Uprate, remains in accordance with the existing FSAR.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.5.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1 "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3 "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4 "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"

9. ESR 99-00217 Revision 0, "Install Replacement Sea-Cure Tubing for the Main Condenser".

10. Appendix A to 10CFR50, "General Design Criteria for Nuclear Power Plants" Criterion 4, "Environmental and Dynamic Effects Design Bases"

2.6 Component Cooling Water System

The Harris Nuclear Plant (HNP) Component Cooling Water System (CCWS) has been evaluated to determine its performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt. This evaluation takes into account the bounding heat loads in Spent Fuel Pools A/B and activation of Spent Fuel Pools C/D with a heat load of 1 MBTU/Hour.

2.6.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the CCWS serves as an intermediate closed cooling water system between radioactive or potentially radioactive systems and the non-radioactive service water system. The CCWS rejects its heat load to the Service Water System. This arrangement minimizes the possibility of leakage of radioactive material into the environment.

The CCWS consists of two 100% CCW heat exchangers, three 100% CCW pumps (one as an installed spare), a component cooling water surge tank, and associated piping, valves, and instrumentation. There are two essential cooling loops, and one nonessential cooling loop. The CCWS is designed to provide effective cooling of plant components, with one train operation, during all operating modes. When two CCWS trains are in operation, one train is aligned to its respective essential loop Residual Heat Removal (RHR) and to the non-essential loop. The second train is aligned to its respective essential loop only. During this alignment, RHR heat load is not shared equally between the two trains, and CCW flow will differ between Train 'A' and Train 'B'.

The existing CCWS maximum supply temperature is 105°F during normal operation and 120°F during shutdown.

The SGR/Uprate does not change existing CCWS design functions.

<u>Configuration Changes</u>:

The CCW pump impellers will be changed to provide increased system flow and cooling capability. This increased cooling capability will accommodate the increased heat loads that result from SGR as well as the activation of Spent Fuel Pool heat from Pools C/D. It will also accommodate the anticipated heat loads from PUR. Flow instrumentation in the CCW System is being evaluated for the increased system flow rates.

Revised Process Conditions:

The CCW system heat load is increased for various operating modes due to the additional heat rejection primarily from the NSSS systems.

The CCW system flow rate will be increased to provide additional CCWS heat exchange capability so that the additional heat load removal requirements from the NSSS systems will be met.

The CCW supply temperatures to NSSS components are conservatively analyzed at slightly higher values as a result of the additional CCWS heat loads.

The CCW system normal and peak operating pressures are increased as a result of the increased CCWS pump performance capability needed for the increased system flow requirements.

The impact of increased heat loads, system flow, supply temperature, and operating pressures on the CCWS under the SGR/Uprate is addressed in Section 2.6.2.

2.6.2 Description of Analyses and Evaluations

The CCWS was evaluated to ensure that, following the SGR/Uprate, the system remains capable of performing required functions in accordance with the existing licensing bases specified in the FSAR.

System performance was evaluated based on the increased heat rejection from the NSSS systems. The hydraulic analysis provides the CCW system flow capacity based on the new CCW pump impeller for the various operating modes. The modes evaluated are listed in table 2.6.1. The NSSS analysis (refer to NSSS Report Section 4.1.6) provides the CCW heat exchanger heat removal rates based on the CCW flow rates and determines the CCW supply temperatures and plant cool down time. The Spent Fuel Pool heat loads used in the respective cases are listed in BOP LR Section 2.9. Based on the analysis, the most limiting operating case occurs during normal shutdown at 350°F Reactor Coolant System temperature.

Table 2.6.1

Evaluated Operating Modes

Plant Start-Up @ 350°F
Plant Start-Up @ 557°F
Normal Operation
Shutdown @ 350°F
Shutdown @ 140°F
Safety Injection
Recirculation @ 243.5°F
Recirculation @ 200°F
Loss of Off-Site Power
Refueling – In-Core Shuffle
Refueling – Normal Full Core Offload
Refueling – Abnormal Full Core Offload

The analyses demonstrate the acceptable performance of the CCWS.

Based on CCWS evaluation results, the CCW supply temperature peaks at 106.8°F during normal operating conditions, and 124.8°F during shutdown. The RCS cooldown analysis assumes that the CCW maximum supply temperature is limited to 120°F (refer to NSSS LR Section 4.1.4) by limiting the RCS cooldown rate. During post-loss of coolant accident (LOCA) recirculation, the CCW supply temperature could peak at 134.2°F. This is acceptable since it is below the system design temperature of 200°F.

The impact of the increase in system flow and operating pressures as a result of the new impeller is addressed as a part of the CCW pumps modification.

CCW Pumps

The existing CCWS pumps (rated for 8,050 gpm at 211 ft) are not capable of providing the system flow required. Therefore, a new impeller design (rated for 10,500 gpm at 237 ft) capable of the increased cooling water flow requirements was evaluated.

The predicted operating range for the new impeller is 63.4% to 113.4% of Best Efficiency Point (BEP) 10,500 gpm (6,658 gpm to 11,907 gpm) compared with the preferred range of 70% to 120%. The low flow (6,658 gpm) is predicted only during the initial Safety Injection operating mode, which has a short duration during a low probability event. The next lowest predicted flow occurs during normal operation and is 75.8% of BEP (7,957 gpm). Therefore, the new impeller design is considered acceptable for the predicted operating range.

The performance curve for the new impeller has a predicted tolerance of ± 1% for BEP, and ± 3% on flow. The CCWS system analysis incorporates a 10% margin for flow to accommodate impeller degradation and instrument errors, and a 6% margin for the software program hydraulic uncertainty. On this basis, there is reasonable assurance that the final, as tested, performance curve will be acceptable for the SGR/Uprate.

The available NPSH of 50 ft exceeds the required NPSH of the replacement impeller over the predicted operating range, of 16 to 32 feet.

CCW Heat Exchangers

The thermal and hydraulic performance capability of the CCW heat exchangers and other coolers serviced by the CCWS are addressed in NSSS LR Section 5.9, NSSS Auxiliary Equipment.

Surge Tank

The component cooling surge tank provides the ability to accommodate changes in CCWS water volume due to changes in operating temperature. As a result of increased heat load on the CCWS, the operating temperature will increase slightly. The resulting operating temperature increases are well within the design temperature of the system, therefore, the SGR/Uprate will have negligible impact on the component cooling surge tank.

Pipe Size

The individual CCWS pipe sections may experience a flow increase of nearly 50% during certain operating conditions resulting in pipe velocities near 20 to 25 ft/sec. However, this occurs during relatively short term operating modes (start-up, shutdown, accident and/or refueling) and is within an acceptable range for the piping. The resultant pressure drops are also within the capability of the proposed performance of the CCW pumps. Pipe erosion effects due to the SGR/Uprate are considered negligible. CCWS flows and velocities during normal plant operation, are within the current design of the system. Therefore, the CCWS pipe sizes are adequate to support the SGR/Uprate.

Pipe Wall Thickness

With the installation of the new design impellers for the CCW pumps, maximum system operating pressures are expected to increase by 5 to 10 psi, but do not exceed the current system design pressure of 150 psig for the base system and 190 psig for portions of the systems. Therefore, the existing Component Cooling Water System pipe wall thickness is adequate for the SGR/Uprate.

Relief Valves

Relief valves or a flow orifice in series with a check valve are used to provide thermal relief protection. The relief protection has been evaluated and found to continue to be acceptable.

The thermal relief valves on the components located outside Containment Building were replaced with a combination locked open valve, check valve and flow orifice. The arrangement eliminated the potential for an inadvertent thermal relief valve actuation while providing thermal protection for each component. The arrangement is unaffected by the SGR/Uprate.

Instrumentation and Controls

The increased CCW system flow rate exceeds the current range for several flow instruments. Evaluations are in process to ensure that the CCW instrumentation will accommodate the increased system flow requirements due to the combined effects of the SGR/Uprate. Review of the scaling and setpoints for the CCW tank level instruments indicates there is no need for any changes to the level instruments.

The results of the evaluation of the CCWS and the individual components to satisfy applicable design and licensing bases in accordance with the FSAR are presented in Section 2.6.4. Acceptance criteria, relevant to the CCWS, are identified in Section 2.6.3.

2.6.3 Acceptance Criteria

The acceptance criteria applicable to the CCW system for SGR/Uprate are:

1) The CCW System supply sufficient cooling water flow rates to the system heat loads and that water supply temperatures remain within acceptable limits.

2) The changes to the CCW system performance due to the new impeller do not adversely impact the integrity of the system, the ability to monitor system performance and the capability of the onsite electrical distribution to support the CCW pump operation.

2.6.4 Results

The SGR/Uprate does not change the CCWS functional design requirements or adversely affect functions. However, configuration changes to the CCWS are required.

The CCWS components were evaluated to determine their capability to meet all functional requirements. The CCW pump impellers require replacement to increase the system cooling capability.

The increased CCW system flow rate exceeds the range for several flow instruments. The instrumentation is being evaluated to assure appropriate range exists to accommodate the increased system flow requirements. Review of the scaling and setpoints for the CCW tank level instruments indicates there is no need for any changes to the level instruments.

With the exception of the pump impeller and instumentation changes, the results of the evaluation support that all components of the CCWS are adequate.

The estimated required BHP of the proposed impeller over the predicted operating range, is 630 to 740, which is within the capability of the current CCW pump motors, which have a nameplate horsepower rating of 800. The estimated BHP is also within the current values used in the electrical load calculations.

2.6.5 Conclusions

As described the CCWS design will be adequate for the SGR/Uprate following the modification to the pump impeller. Process conditions such as pipe velocity, CCWS operating temperature and maximum pressures will increase as a result of SGR/Uprate. However, these values remain within the acceptable limits. The revised CCWS heat load calculations indicate that subsequent to completion of the above modification, the CCWS will remain capable of performing all required functions, under SGR/Uprate conditions.

The results obtained with the Delta 75 RSG at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSG at the current NSSS power level of 2787.4 MWt.

2.6.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.7.3 Component Cooling Water System

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

2.7 Normal and Emergency Service Water Systems

The Harris Nuclear Plant (HNP) Normal Service Water (NSW) and Emergency Service Water (ESW) Systems have been evaluated to determine their performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt. This evaluation takes into account the bounding heat loads in Spent Fuel Pools A/B and activation of Spent Fuel Pools C/D with a heat load of 1 MBTU/Hour.

2.7.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the function of the Service Water System (SWS) is to remove heat from the plant auxiliary systems and equipment. The SWS consists of two interconnected subsystems, the Normal Service Water (NSW) System and the Emergency Service Water (ESW) System.

The NSW removes plant heat loads from auxiliary components associated with the power conversion system, reactor operation and miscellaneous building services during normal plant operation. The NSW also functions during start-up, normal shutdown and hot standby. The NSW System circulates water from the Cooling Tower (CT) basin through plant auxiliary components and back to the cooling tower. The NSW serves no safety-related function, but it provides all cooling water requirements to the ESW portion of the loop during normal plant operation including start-up, normal shutdown and hot standby.

The ESW removes essential plant heat loads associated with the reactor auxiliary components for dissipation in the Ultimate Heat Sink (UHS) during emergency operation. The ESW System circulates water from the UHS through reactor auxiliary components required for safe shutdown of the reactor following an accident, and returns the water to the UHS. The ESW System also performs its cooling function during Loss of Offsite Power (LOOP) events. The ESW is arranged into two completely independent redundant trains, each capable of supplying sufficient cooling water for plant safety. Upon receipt of a Safety Injection Actuation Signal (SIAS), the ESW loops are automatically isolated from each other and the NSW loop. The ESW loops are then supplied with water from their respective ESW pumps.

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change existing NSW and ESW design functions. The existing SWS remains capable of satisfying regulatory commitments in accordance with the existing FSAR under SGR/Uprate conditions.

Configuration Changes:

There are no configuration changes associated with the SWS under the SGR/Uprate conditions.

Revised Process Conditions:

Heat rejection loads to the SWS and component flow rates have changed.

The impact of changed heat rejection loads and component flow rates on the Service Water System under the SGR/Uprate conditions is addressed in Section 2.7.2.

2.7.2 Description of Analyses and Evaluations

The SWS was evaluated to ensure that, following the SGR/Uprate, the system remains capable of performing required functions in accordance with the existing licensing basis specified in the FSAR.

The SWS analysis showed that heat rejection loads to the SWS and component flow rates have changed in support of the SGR/Uprate. However, the total design flows for the NSW and the ESW remain unchanged under SGR/Uprate conditions.

The following changed heat rejection loads and component flow rates were addressed:

The ability of the cooling tower to maintain a 95°F circulating water temperature for the NSW system was evaluated in BOP LR Section 2.5. The SGR/Uprate heat loads were bounded and the service water temperature was determined to be 95.2°F.

Under SGR/Uprate conditions, the service water flow rate through the CCW heat exchanger has been increased from 8,500 gpm to 10,000 gpm during normal operation and from 8,000 gpm to 8,500 gpm during accident conditions when only the essential heat loads are removed. The ESW flow represents the minimum flow associated with the most limiting single active failure. The new flow rates are less than the design flow rate of 12,000 gpm for the heat exchanger and are acceptable with regard to CCW Heat Exchanger design.

During emergency operation, the UHS dissipates the ESW System heat loads. The maximum heat load to UHS includes the NSSS heat loads such as sensible and decay heat and BOP heat loads such as from the containment fan coolers. To ensure that the maximum service water temperature following a LOCA does not exceed 95°F, an analysis was performed assuming a starting temperature of 94°F (Technical Specification Limiting Condition for Operation) and the one day worst meteorological conditions. Based on the analytical results, the maximum temperature in the 24-hr period increases by 0.19°F (95.14°F to 95.33°F).

However, the actual maximum UHS temperature is expected to be below 95°F based on conservative assumptions in the analysis. In particular, the pre-accident meteorological conditions include a 9-day severe temperature condition followed by the worst single day. The resulting pre-accident prediction of the UHS temperature was 94.2°F. Also, if a 1°F temperature rise across the surface of the water is assumed, the bulk temperature decreases by 0.17°F. Since the ESW pumps draw water well below the surface of the

UHS, the temperature of the water supplied to plant systems will remain below 95°F. Because the UHS temperature response was found to be insignificantly affected by the SGR/Uprate, the existing analysis of the maximum evaporation remains valid and there is a sufficient inventory for 30 days. The heat rejected to the ESW following a LOCA has increased, however, the resulting increase in UHS temperature is negligible. Therefore, the UHS capability is not changed as a result of the SGR/Uprate.

The minimum Auxiliary Feedwater (AFW) System flow to intact Steam Generators in FSAR Chapter 15 events is 390 gpm. This flow is used as the demand that would be placed on the ESW System if the AFW supply were manually switched from the Condensate Storage Tank to the ESW System. The increase in the service water requirement from the previous minimum of 380 gpm is within the existing flow capability of the ESW System.

2.7.3 Acceptance Criteria

The licensing basis for the SWS is described in the FSAR. The acceptance criteria for SWS relevant to SGR/Uprate concerns the SWS function to supply cooling water to various non-safety related and safety-related components and heat exchangers at a specified temperature during normal and accident plant operation.

The Emergency Service Water System is required to supply adequate flow to safety related heat exchangers for the following:

1. CCW Heat Exchangers (RCS Cooldown and LOCA Long Term Mass and Energy Release)
2. Containment Fan Coolers (Containment Analysis)
3. Auxiliary Heat Loads (Emergency Diesel Jacket Water Heat Exchanger, Essential Chilled Water Chillers, Charging Safety injection Pump Oil Coolers, etc.)
4. Maintain maximum ESW supply temperature less than or equal to 95°F.

The design, operation, and functional capabilities of the SWS and the Ultimate Heat Sink described in the FSAR, and as affected by the SGR/Uprate, were evaluated against the acceptance criteria. The results of these evaluations are described in Section 2.7.4.

2.7.4 Results

The SGR/Uprate does not change the NSW and ESW system designs, or adversely affect system functions. As stated in Section 2.7.1, the SGR/Uprate requires no NSW or ESW System configuration changes.

Process condition changes involving the systems are within existing system capabilities. The existing NSW and ESW Systems design pressures and temperatures and maximum operating pressures and temperatures are not changed as a result of the SGR/Uprate.

The performance of the ESW system in support of other analysis is as follows:

1 The performance of the removal of heat from the containment via the RHR and CCW heat exchangers is accounted for in the Long Term Mass and Energy release described in NSSS LR Section 6.4.1.
2 The performance of the ESW system is an input to the Containment Fan Cooler performance assumed in Containment analysis. The results of the Containment analyses are described in BOP LR Section 2.24.
3 The performance of the ESW system is an input to the RCS cooldown analysis. The results of this analysis is described in NSSS LR Section 4.1.4.
4 The analysis presented in BOP LR Section 2.7.2 shows that the UHS can maintain the maximum supply temperature less than or equal to 95°F.

2.7.5 Conclusions

The existing NSW and ESW system designs are adequate for the SGR/Uprate. There are no configuration changes to the systems. The existing system components are adequate to meet the SGR/Uprate conditions. NSW and ESW heat removal functions and capabilities are not adversely affected.

The results obtained with the Delta 75 RSG at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSG at the current NSSS power level of 2787.4 MWt.

2.7.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.7.4 Emergency Service Water System
   3/4.7.5 Ultimate Heat Sink

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

2.8 Auxiliary Feedwater System

The Harris Nuclear Plant (HNP) Auxiliary Feedwater System (AFW) has been evaluated to determine its performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.8.1 Introduction and Background

The function of the AFW is to supply feedwater to the secondary side of the steam generators (SGs) at times when the normal feedwater system is not available, thereby maintaining the heat sink capabilities of the SGs. This system functions during startup, hot standby, and cooldown and during plant transients.

During plant transients, the AFW is relied upon to prevent core damage resulting from a loss of normal feedwater flow. These transients include main feedwater line break, main steam line break, steam generator tube rupture (SGTR), loss of coolant accident (LOCA) and/or loss of off-site power (LOOP). The AFW provides feedwater to the unaffected steam generators to maintain their inherent heat sink capability.

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change auxiliary feedwater system design functions. The SGR/Uprate involves configuration changes and revised process conditions to the AFW, as described below.

Configuration Changes:
The AFW system piping is being modified due to a change in AFW nozzle orientation of the RSG.

Revised Process Conditions:
The SGR/Uprate has resulted in negligible changes in the system resistance along with a reduction in required AFW flow from 400 gpm to 390 gpm. The 390 gpm minimum flow is evaluated for various scenarios of total flow to 2 or 3 SGs respective of the limiting scenario.

2.8.2 Description of Analyses and Evaluations

The AFW was evaluated to ensure that, following the SGR/Uprate, the system is capable of performing required functions in accordance with the design bases requirements (see table 2.8.2-1 and 2.8.2-2), except as noted.

Analysis and evaluations were performed to ensure that the system would adequately perform its design function during the required plant conditions. AFW system components were evaluated with respect to the SGR/Uprate conditions and were found to be capable of performing their design function. The required transients were evaluated for bounding conditions and single active failures as appropriate to ensure that the minimum or maximum flow and functional requirements are maintained. System response was evaluated for:

- Loss of normal Feedwater with and without offsite power;
- Feedwater Line Break (FWLB)-See note below

- Main Steam Line Break (MSLB);
- Station Blackout (loss of all AC power)
- Loss of Coolant Accident (LOCA);
- Steam Generator Tube Rupture (SGTR)

NOTE: With the changes in design characteristics in the replacement SG, additional analysis is required to ensure adequate AFW system response to a Feedwater Line Break (FWLB) is maintained. In a FWLB, the turbine driven AFW pump will receive an automatic start signal. Analysis to support minimum pump and flow assumptions utilized in FSAR Chapter 15 events has been completed, but design bases response to this specific event has not. The results of this analysis are incomplete at this time. Results will be submitted upon completion of the analysis.

AFW system pump capabilities were developed based on the existing pump capabilities (pump curves) for both the motor and turbine driven pumps. A minimum available (degraded) pump curve was developed for both pumps based on the establishment of curves that intersect the current Technical Specification surveillance requirements at both low and high flow conditions. These curves were then utilized to predict system performance at the various conditions required in the plant analysis. Where a stronger pump yielded more restrictive results, the nominal (existing) pump curves were used. Where a weaker pump would be more conservative, the minimum available curves were utilized

The Station Blackout (SBO) transient has also been reviewed to determine if there is sufficient inventory in the Condensate Storage Tank (CST) for the AFW System to mitigate the accident.

The Condensate Storage Tank (CST) was evaluated with respect to minimum CST useable inventory requirements and SGR/Uprate impact. Under SGR/Uprate conditions, the minimum useable CST inventory has increased based on the re-analysis, using the current Technical Specifications bases requirement, to a value greater than the safety related dedicated inventory.

A change to the Technical Specifications Bases (Bases 3/4.7.1.3 "Condensate Storage Tank") is proposed to reduce the time at hotstandby from 12 hours to 6 hours, which is greater than the 4 hours identified in the Standard Review Plan Section 10.4.9 (NUREG 0800). This change conservatively assumes a 6 hour cooldown period to an RCS temperature of 325°F (RHR Cut-in temperature is 350°F), which includes a 1.4 hour allowance for aligning the RHR system. This change will enable a reduction in the required dedicated CST inventory to less than that currently available, and will not require a plant modification.

2.8.2.1 Minimum AFW Flow Requirements – Chapter 15 Accident Analyses, Table 2.8.2-1

There are eight (8) minimum flow transient cases identified in Table 2.8.2-1 that depend on AFW supply. Transients I through VII were evaluated using one pump to ensure that the minimum flow would be supplied for all cases of steam generator pressures, intact flow paths, and degraded pumps. The evaluations focused on the transients with the most limiting boundary conditions. Transient VIII is for a Steam Generator Tube Rupture Event, with three pumps in operation, and was evaluated independently.

2.8.2.2 Maximum AFW Flow Evaluation – Chapter 15 Accident Analyses, Table 2.8.2-2

There are two (2) maximum transient cases identified in Table 2.8.2-2 that depend on AFW supply.

The results of the evaluation of the AFW, and its individual components, to satisfy applicable design and licensing bases in accordance with the FSAR is presented in Section 2.8.4. Acceptance criteria, relevant to the AFW, are identified in Section 2.8.3.

Table 2.8.2-1
Minimum AFW Flow Requirements – Chapter 15 Accident Analyses Requirements (Sheet 1 of 1)

Transient No	Chapter 15 Transient Event	No of Pumps Operating	SG Pres. (psig)/ No. of SGs Available	AFW Total Flow Required, gpm Pre-SGR/U	Post-SGR/U	Chapter 15 Assumptions	Remarks
I	Loss of MFW with LOOP FSAR Section 15.2.6	1 MD	1217/2	400	390	Note 1	
II	Loss of MFW with LOOP Available 15.2.7	1 MD	1217/3	400	390	Note 2	
III	Feed Line Break 15.2.8	1 MD	1217/2	400	390	Note 1	With Loss of Offsite Power
IV	Feed Line Break 15.2.8	1 MD	12171/2	400	390	Note 1	With Offsite Power Available
V	SBLOCA 15.6.5	1 MD	1217/3	400	390	Note 2	
VI	MSLB 15.1.5	1 MD	1217/2	400	390	Note 1	After faulted SG is isolated
VII	SGTR 15.6.3	1 MD	1106/2	400	390	Note 1	After faulted SG is isolated
VIII	SGTR 15.6.3	2 MD + 1 TD	1106/3	1200	1170	Note 3	Before faulted SG is isolated (Underfill Case)

<u>Chapter 15 Assumptions</u>

Note 1: AFW delivers 390 gpm to two (2) SGs using one pump. For conservatism assume MD AFW pump with lower capacity is available

Note 2: AFW delivers 390 gpm to three (3) SGs using one pump. For conservatism assume MD AFW pump with lower capacity is available

Note 3: AFW delivers a minimum flow of 1170 gpm using all three (3) AFW pumps. It is further assumed that the flow to each SG is at least equal to 390 gpm.

Table 2.8.2-2
Maximum AFW Flow Requirements – Chapter 15 Accident Analyses Requirements (Sheet 1 of 1)

Transient No	Chapter 15 Transient Event	No of Pumps Operating	SG Pres. (psig)/ No. of SGs Available	AFW Total Flow Required, gpm Pre-SGR/U	AFW Total Flow Required, gpm Post SGR/U	Chapter 15 Assumptions	Remarks
I	MSLB 15.1.5	2 MD + 1 TD	Note 1/1	2600	3000	Note 2	Before faulted SG is isolated
II	SGTR 15.6.3	2 MD + 1 TD	1106/3	1500	1500	Note 3	Before faulted SG is isolated (Overfill Case)

Maximum SG Pressure

Note 1: Lowest SG pressure: Pre-SGR/Uprate, 270 psia
Post-SGR/Uprate, 289.2 psia

Chapter 15 Assumptions

Note 2: Assumed maximum AFW flow to faulted SG is not greater than 3000 gpm until faulted SG is isolated. For conservatism assume that all three pumps flowing to faulted steam generator (maximum blowdown energy release data)

Note 3: Assume maximum flow of all three pumps does not exceed 1500 gpm. It is further assumed that the flow per steam generator does not exceed 500 gpm.

2.8.3 Acceptance Criteria

The design bases requirements for the system are described in section 10.4.9.1 of the FSAR. The design bases requirements are met if minimum flow requirements as are noted in Tables 2.8.2-1 & 2 are met for the respective scenarios.

2.8.4 Results

The SGR/Uprate does not change the AFW design function. Configuration changes required for the AFW under SGR/Uprate, and any changes to process conditions, are either within the capabilities of the existing design, or a change in the current licensing basis will be requested.

CST Licensing Change

The Technical Specifications were evaluated with consideration given to the SGR/Uprate. In the bases for Condensate Storage Tank (CST) Technical Specification (TS) 3/4.7.1.3, the duration of the hot standby period will be changed from the current 12 hours to 6 hours. This change will preclude an increase in minimum CST inventory that would otherwise be necessary to accommodate increased heat loads during hot standby. The TS B3/4.7.1.2 will also be revised. The design of the AFW system must ensure that the plant can remain in hot standby for 6 hours and then cooldown the Reactor Coolant System to less than 325°F, in lieu of 350°F, before the residual heat removal system is utilized for long term plant cooldown.

The shorter Hot Standby duration is within the Standard Review Plan (NUREG 0800) requirements. According to NUREG 0800, Section 10.4.9, Item I.18, the system design shall have the capability to permit operation at hot shutdown for at least 4 hours followed by cooldown to the RHR cut-in temperature assuming the worst single active failure in accordance with Branch Technical Position RSB 5-1.

AFW Pump Technical Specification Bases Change

The minimum AFW pump surveillance performance requirements are currently based on a pump degradation of 4% from the factory pump curves. This is being changed to delete a reference to specific percentage degradation. The analysis were run using pump curves which intercepted the current surveillance testing points at both the low and high flow conditions. Thus a specific degradation value is not necessary for the bases for this specification.

The AFW pumps are adequately designed to comply with all system design basis requirements, except as noted above. Analysis results for the additional FWLB scenario will be presented in a subsequent submittal.

Review of the SBO transient has determined that the minimum required inventory to satisfy the cooldown operation is 116,178 gallons and the available inventory is 238,000 gallons in the CST. The CST capacity satisfies the SBO requirements for SGR/Uprate.

The Anticipated Transient Without Scram (ATWS) analysis results indicate that the AFW design has sufficient capacity to cope with the ATWS conditions for the SGR/Uprate. Specific results are noted in the following table.

The AFW System was analyzed for the FSAR Chapter 15 events identified in tables 2.8.2-1 and 2.8.2-2. The following table identifies the calculated flows for the respective events, which were determined by the analysis conducted on the system. The flows identified are for the most restrictive pump or pumping combination for the respective scenario:

Transient	Chapter 15 Event	Pumps	AFW Flow Required	Calculated AFW Flow
Minimum Flow Cases				
I	Loss of MFW with LOOP FSAR Section 15.2.6	1 MD-Degraded Pump to 2 SG	390 gpm	396 gpm
II	Loss of MFW with LOOP Available 15.2.7	1 MD- Degraded Pump to 3 SG	390 gpm	401 gpm
III	Feed Line Break 15.2.8 (w/LOOP)	1 MD Degraded Pump to 2 SG	390 gpm	396 gpm
IV	Feed Line Break 15.2.8	1 MD- Degraded Pump to2 SG	390 gpm	396 gpm
V	SBLOCA 15.6.5	1 MD- Degraded Pump to 3 SG	390 gpm	401 gpm
VI	MSLB 15.1.5	1 MD- Degraded Pump to 2 SG	390 gpm	396 gpm
VII	SGTR 15.6.3	1 MD- Degraded Pump to 2 SG	390 gpm	396 gpm
VIII	SGTR 15.6.3	2 MD + 1 TD to 3 SG	1170 gpm	~1350 gpm *
ATWS		2 MD + 1 TD to 3 SG	1400 gpm	1798 gpm
Maximum Flow Cases				
I	MSLB 15.1.5	2 MD + 1 TD	<3000 gpm	2540 gpm
II	SGTR 15.6.3	2 MD + 1 TD	<1500 gpm	~1350 gpm *

*Previous system testing was performed to determine the values presented in the SGTR cases. HNP typically performs this test after system resistance changes to confirm TD pump speed control settings and ensure that the flow requirements as stated above are maintained. Since piping resistance changes are small, and the SG pressures for the events do not change, little if any change to the controller setting is expected.

2.8.5 Conclusions

The existing AFW design is adequate for the SGR/Uprate. The AFW piping will be modified to accommodate RSGs. Additional analysis is required for AFW response to a FWLB.

The results obtained with the Delta 75 RSG at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSG at the current NSSS power level of 2787.4 MWt.

2.8.6 References

1. HNP Final Safety Analysis Report Section 10.4.9

2. HNP Technical Specifications
   3/4.7.1.2 Auxiliary Feedwater System
   B3/4.7.1.2 Auxiliary Feedwater System
   3/4.7.1.3 Condensate Storage Tank
   B3/4.7.1.3 Condensate Storage Tank

2.9 Fuel Pools

The Harris Nuclear Plant (HNP) A/B and C/D Fuel Pool Cooling & Cleanup Systems (FPCCS) have been evaluated to determine performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs). This evaluation takes into account the bounding heat loads in Spent Fuel Pools A/B and activation of Spent Fuel Pools C/D with a heat load of 1 MBTU/Hour.

2.9.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the function of the FPCCS is to provide cooling to both the new fuel and spent fuel pools while maintaining the water quality in the pools. The FPCCS is comprised of three subsystems: two redundant cooling loops, two cleanup loops, and a skimmer loop.

The cooling loops are each capable of removing 100% of the expected decay heat load from spent fuel stored in the Spent Fuel Pools (SFP). Each train consists of a cooling pump, a heat exchanger, and a strainer. To provide redundancy each train is connected to different power sources and if there is a complete loss of offsite power may also be connected to the emergency diesel generators. The heat exchangers are of the shell and straight tube type with the water from the fuel pools running through the tubes. The heat is transferred to the Component Cooling Water (CCW) System on the shell side. Administrative controls are placed on the minimum cooling time prior to transferring irradiated fuel from the core to the storage facility in order to maintain the pool at less than or equal to 137°F for the core off load cases. As described in the FSAR for a single failure to one of the FPCCS loops, the remaining FPCCS loop is capable of maintaining the SFP at or below 137°F.

The worst situation involved Emergency Core Cooling System (ECCS) recirculation from the containment sump when the CCW system was isolated from the nonessential header and in turn the FPCCS heat exchangers. At 5.56 hours from the time of LOCA initiation, the heat load on the RHR heat exchanger will be low enough to permit realignment of CCW to the Spent Fuel Pool heat exchanger. The Spent Fuel Pools A/B will heat up to 137°F in 5.56 hours, assuming an initial temperature of 112.7°F and a normal maximum heat load subsequent to a LOCA of 16.84 MBTU/hr. With this heat load, 2.97 hours is available for manual actions to restore CCW to the spent fuel pool heat exchanger prior to reaching 150°F in the pools. The heatup rate of Spent Fuel Pools A/B pools bounds that of Spent Fuel Pools C/D.

The cleanup loops are provided to remove fission and corrosion products, which may enter the SFP water from leaking fuel assemblies or as a result of the fuel transfer process itself (e.g. - mixing of water from other sources), as well as other contaminants. The performance of the demineralizer is also monitored and the resin replaced when the ion exchange media is depleted.

The Steam Generator Replacement (SGR) does not change existing FPCCS design functions.

Configuration Changes:

There are no configuration changes associated with the FPCCS under the SGR.

Process Conditions:

For SGR there are no revisions to the Spent Fuel decay heat loads. The heat loads for the Spent Fuel Pools A/B are as listed in the FSAR Table 9.1.3-2. The pending License Amendment application for Spent Fuel Pool C/D activation will limit heat load to a maximum of 1.0 MBTU/hr.

2.9.2 Description of Analyses and Evaluations

The FPCCS were evaluated to determine if, following SGR, the systems remain capable of performing required functions in accordance with the existing licensing bases specified in the FSAR.

The following table summarizes the Spent Fuel Pools A/B and (proposed) Spent Fuel Pool C/D heat loads. As described in FSAR Section 9.1.3.3, administrative controls are placed on the cooling time prior to transfer of irradiated fuel to the storage facility to maintain the pools at less than or equal to 137°F during core off-load cases.

Table 2.9.1
Spent Fuel Pool Heat Duty in MBTU/hr

Basis or Operating Conditions/Cases	**Pre-PUR**	
	A/B Pools	**C/D Pools***
Heat load in Pools:		
Normal Case Normal Incore shuffle	16.84	1.0
Maximum Normal Case Full Core Offload	35.06	1.0
Abnormal Case Post Outage Full Core Offload [approximately 30 days after Normal Incore Shuffle]	35.87	1.0
Normal Operation Normal Operating Condition [Prior to plant start-up]	16.84[1]	1.0
Safe Shutdown and LOCA Conditions Safe Shutdown Recirculation Phase - DBA LOCA	16.84[1]	1.0

*Proposed heat load from pending License Amendment application to place Spent Fuel Pools C/D in service.

[1]The value conservatively bounds the decay heat present when the reactor is restarted and reaches equilibrium core decay heat conditions.

The maximum equilibrium temperatures are ≤140°F for Spent Fuel Pools A/B and Spent Fuel Pools C/D assuming a CCW Supply Temperature of 120 °F.

The pool bulk equilibrium temperatures during RCS cooldown were analyzed based on the CCW flow balance, CCW supply temperature, and the SGR and the heat loads listed in Table 2.9.1.

FSAR Table 9.1.3-2 provides the heat up rates for Spent Fuel Pools A/B in the event that fuel pool cooling stops. Based on Table 2.9.1, the Spent Fuel Pools C/D heatup rates are bounded by the Spent Fuel Pools A/B heat up rates.

During the recirculation phase of the LOCA, when the Emergency Core Cooling System is aligned to recirculate from the containment sump to the Reactor Coolant System, the CCW flow to the Spent Fuel Pool Heat Exchanger is isolated. The CCW flow is assumed to be restored to FPCCS within approximately 8.5 hours.

For SGR, the discussion in FSAR Section 9.1.3.3 is unchanged and remains valid for restoration of spent fuel pool cooling following LOCA.

Fuel Pool Cooling and Clean-up Pumps

The existing 100% capacity Fuel Pool Cooling pump provided in each redundant loop of the two independent FPCCS for Spent Fuel Pools A/B and Spent Fuel Pools C/D respectively are adequate for the SGR and the proposed Spent Fuel Pool C/D activation.

Fuel Pool Heat Exchangers

One 100% capacity fuel pool heat exchanger is provided in each redundant loop of the two independent FPCCS for Spent Fuel Pools A/B and Spent Fuel Pools C/D respectively. Component Cooling Water circulates through the shell, while the fuel pool water circulates through the tubes. Each fuel pool heat exchanger has a design heat duty of 15.06×10^6 BTU/hr.

2.9.3 Acceptance Criteria

The licensing basis for FPCCS are described in the FSAR. Relative to the SGR the relevant acceptance criteria is:

The function of the FPCCS is to maintain the quality of the Spent Fuel Pool water while removing decay heat from spent fuel assemblies stored in the Spent Fuel Pools, and maintaining a bulk water temperature of less than or equal to 140°F.

The design, operation and functional capabilities of the FPCCS described in the FSAR, and as affected by the SGR and C/D fuel pool activation, were evaluated against the acceptance criteria. The results of these evaluations are described in Section 2.9.4.

2.9.4 Results

The SGR does not change the system and component functions. The results of the Post-LOCA Spent Fuel Pool heatup remain unchanged because the heat load is unchanged.

The current design conditions remain unaffected due to the SGR. No instrumentation or control changes are required for the FPCCS as a result of the SGR.

2.9.5 Conclusions

The existing FPCCS design is adequate to support the SGR. There are no required configuration changes to the system design. The existing system components are adequate to meet the SGR conditions.

2.9.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.9.11 Water Level - New and Spent Fuel Pools

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2)," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

2.10 Chemical and Volume Control System

The Harris Nuclear Plant (HNP) Chemical and Volume Control System (CVCS) (including the Boron Recovery System) has been evaluated to determine its performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.10.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the CVCS is designed to provide the following functions in support of the Reactor Coolant System (RCS):

- Maintain a programmed level in the pressurizer (i.e., maintain required water inventory in the RCS).
- Maintain seal-water injection flow and bearing cooling to the reactor coolant pumps (RCPs).
- Control reactor coolant water chemistry conditions, activity level, soluble chemical neutron absorber concentration and makeup.
- Provide injection flow to the RCS following actuation of the Emergency Core Cooling System (ECCS). Note that portions of the CVCS are shared with the ECCS, providing a safety injection function. This function is addressed in BOP LR Section 2.12 "Safety Injection System."
- Provide a means for filling, draining and pressure testing of the RCS.

The CVCS consists of several subsystems: the Charging, Letdown and Seal Water System; the Reactor Coolant Purification and Chemistry Control System; the Reactor Makeup Control System; and the Boron Thermal Regeneration System. The CVCS operates in conjunction with the Boron Recycle System (BRS).

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change CVCS design functions. The CVCS remains capable of satisfying regulatory commitments in accordance with the existing FSAR.

Configuration Changes:

There are no configuration changes associated with the CVCS under the SGR/Uprate conditions.

Revised Process Conditions:

The revised process conditions are increased RCS maximum operating temperatures, and pressures, volume and changes to system flow rates. Changes to the process conditions are minor and within the capabilities of the existing design.

2.10.2 Description of Analyses and Evaluations

The CVCS was evaluated to ensure that, following the SGR/Uprate, the system remains capable of performing required functions in accordance with the existing licensing bases specified in the FSAR.

Following the SGR/Uprate there are certain process condition changes that will affect the CVCS. These revised process conditions, including increased RCS maximum operating temperatures, pressures, volume and changes to system flow rates, have been determined to be within the existing CVCS design capabilities (e.g., see Section 5.9, "NSSS Auxiliary Equipment").

Regarding pipe stress analysis, the original pipe stress analyses were performed using the Ebasco computer program, PIPESTRESS 2010. For any revised computer stress analysis included in this report the CP&L computer program, ADLPIPE, was used

Adequate charging flow is provided by the existing CVCS design through the tube-side of the heat exchanger for the various SGR/Uprate letdown flow rates to prevent the Regenerative Heat Exchanger outlet (shell side) temperature from exceeding the 380°F high temperature alarm setpoint.

The SGR/Uprate project results in an increased RCS volume. This increased volume is well within the boration capabilities of the CVCS. In addition, the increased volume will make an uncontrolled boron dilution event less of an issue due to additional time available to mitigate such an event before loss of shutdown margin.

Following the SGR/Uprate, the CVCS remains capable of:

- Maintaining required water inventory in the RCS,
- Maintaining reactor coolant water chemistry conditions, activity level, soluble chemical neutron absorber, makeup and RCP seal injection and bearing cooling,
- Providing injection flow to the RCS following actuation of emergency core cooling, and
- Providing a means for filling, draining and pressure testing the RCS.

Therefore, the CVCS functions and capabilities are not adversely affected and remain in accordance with the existing FSAR.

The results of the evaluation of the CVCS's ability to satisfy applicable design and licensing bases in accordance with the FSAR under SGR/Uprate conditions are presented in Section 2.10.4. Acceptance criteria, relevant to the CVCS, are identified in Section 2.10.3.

2.10.3 Acceptance Criteria

The licensing bases for the HNP CVCS are described in the FSAR. The design and licensing bases of the plant require that the CVCS be capable of borating the RCS at a rate sufficient to match the maximum rate of reactivity insertion that occurs due to the decay of xenon following an extended shutdown from full power at any point in core life. Sufficient charging flow is required to prevent the Regenerative Heat Exchanger outlet (shell-side) temperature from exceeding the 380°F high temperature alarm setpoint.

The primary functions of the CVCS are to maintain RCS water inventory, boron concentration, and water chemistry. To perform these functions, the CVCS must meet the following requirements: (1) the parts of the system that constitute the reactor coolant pressure boundary can withstand the expected RCS conditions, (2) boration meets the design requirements for reactivity control, and (3) with the exception of the RCP seal injection line, the system can be automatically isolated during all events requiring its isolation.

Regulatory Guides (RG) are used as guidance (refer to FSAR Section 1.8 for CP&L's compliance to each RG).

The design, operation, and functional capabilities of the CVCS described in the FSAR, and as affected by the SGR/Uprate, were evaluated against the acceptance criteria. The results of these evaluations are described in Section 2.10.4.

2.10.4 Results

The SGR/Uprate does not change the CVCS, or associated functions, as described in the FSAR. As described in Section 2.10.1, there are no configuration changes required for the CVCS under SGR/Uprate conditions. Any changes to CVCS process conditions are minor and within the capabilities of the existing design. Therefore, operation of the CVCS under the SGR/Uprate conditions does not adversely impact system functions. Additionally, the minor process condition changes do not affect the isolation capability of the CVCS.

2.10.5 Conclusions

The existing CVCS design is adequate for the SGR/Uprate. There are no configuration changes to the system design. The CVCS is capable of maintaining required water inventory in the Reactor Coolant System. The CVCS is capable of maintaining reactor coolant water chemistry conditions, activity level, soluble chemical neutron absorber, makeup and RCP seal injection and bearing cooling. The CVCS is capable of providing injection flow to the RCS following actuation of emergency core cooling. Existing CVCS components are adequate to meet the SGR/Uprate conditions. Therefore, CVCS functions and capabilities under SGR/Uprate conditions are not adversely affected and remain in accordance with the existing FSAR. See NSSS Section 4.1.2 for further discussion of the CVCS.

The results obtained with the Delta 75 RSG at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSG at the current NSSS power level of 2787.4 MWt.

2.10.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.1.1 Boration Control
   3/4.1.2 Boration Systems
   3/4.4.6 Reactor Coolant System Leakage
   3/4.4.7 Chemistry
   3/4.5 Emergency Core Cooling Systems
   3/4.6.3 Containment Isolation Valves

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1" dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"

9. RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste Containing Components of Nuclear Power Plants," (Rev. 3, 02/76)

10. RG 1.29, "Seismic Design Classification," (Rev. 3, 09/78)

11. GL 89-10, "Safety Related Motor-Operated Valve Testing and Surveillance," (Rev. 0, 06/28/89).

12. 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants"
    Criterion 1, "Quality Standards and Records"
    Criterion 2, "Design Bases for Protection Against Natural Phenomena"
    Criterion 14, "Reactor Coolant Pressure Boundary"
    Criterion 29, "Protection Against Anticipated Operational Occurrences"
    Criterion 33, "Reactor Coolant Makeup"
    Criterion 35, "Emergency Core Cooling"
    Criterion 60, "Control of Release of Radioactive Material to the Environment"
    Criterion 61, "Fuel Storage and Handling and Radioactivity Control"

2.11 Residual Heat Removal System

The Harris Nuclear Plant (HNP) Residual Heat Removal System (RHRS) has been evaluated to determine its performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.11.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the function of the RHRS is to remove decay heat from the Reactor Coolant System (RCS) when the plant is not at power operation and when the RCS pressure does not exceed a nominal value of 450 psig. When at shutdown conditions, heat in the RCS is generated by radiological decay of the reactor core, the pressurizer heaters and reactor coolant pump operation. The RHR pumps are also used to transfer water between the Refueling Water Storage Tank (RWST) and the reactor cavity during refueling operations.

During plant start-up, the RHR system continues to operate until the reactor coolant exceeds 325°F and 400 psig, at which time the system may be shutdown by the operator.

During plant shutdown, the RHR pumps and heat exchangers are placed in service to provide additional core cooling when the reactor coolant decreases to about 350°F and a nominal pressure of 450 psig. (Note that the actual set point will be less than 450 psig to allow for instrument error and elevation difference with respect to RHR relief valves). Although both RHR trains are normally used, the use of only one train has no detrimental effect; the cooldown time is extended, however. The operator manually controls the rate of cooling by adjusting the reactor coolant flow through the tube side of the RHR heat exchangers. Total RHR flow is automatically maintained by flow transmitters and flow control valves.

During refueling, operation of the RHR system is continued throughout the refueling process. In addition, the system can be aligned to allow the RHR pumps to be used to transfer water between the RWST and the reactor cavity.

Operation of the RHRS during accidents is addressed in Section 2.12, "Safety Injection System," of the BOP Licensing Report.

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change existing RHRS design functions. The RHRS remains capable of satisfying regulatory commitments in accordance with the existing FSAR under SGR/Uprate conditions.

<u>Configuration Changes</u>:

There are no configuration changes associated with the RHRS under SGR/Uprate conditions.

Revised Process Conditions:

The containment sump temperature, sump water level, and containment pressure are increased following the most limiting design bases accidents for the SGR/Uprate conditions. The decay heat load will increase following SGR/Uprate under both accident and normal shutdown conditions. The impact of higher temperature and water level, and increased decay heat load to the RHRS under the SGR/Uprate conditions is addressed in Section 2.11.2.

2.11.2 Description of Analyses and Evaluations

The RHRS was evaluated to ensure that, following the SGR/Uprate, the system remains capable of performing required functions in accordance with the existing licensing bases specified in the FSAR.

The RHRS is designed to reduce the temperature of the RCS from 350°F to 140°F within 23.3 hours. The heat load handled by the RHRS, during a transient cooldown, includes residual and decay heat from the core along with heat from the reactor coolant pumps. The decay heat load is expected to increase after SGR/Uprate. This increase will lengthen the cooldown duration. The new plant cooldown analysis to support SGR/Uprate confirms that the existing RHRS design has the cooling capacity to bring the reactor to a safe shutdown mode.

The RHR pumps have been evaluated in regard to the adequacy of their net positive suction head (NPSH) to support SGR/Uprate. The most limiting conditions, when the RHR pumps are in recirculation mode drawing from the sump, has been reviewed. The results indicate that the available NPSH of the RHR pumps will support SGR/Uprate.

System relief valves were reviewed to ensure that valve performance was not adversely affected due to SGR/Uprate. The adequacy of the RHR suction relief valves for SGR/Uprate conditions is subject to additional assessment.

A postulated pipe crack in a moderate energy line (i.e., 14" RWST line) has been analyzed to determine the worst flooding effect in the Reactor Auxiliary Building at elevations 190' and 216' under SGR/Uprate conditions. Although there will be a slight increase to the potential flood level under SGR/Uprate conditions compared to the existing design, the difference is very small (i.e., 0.7 inches) and there is no impact on plant safety-related equipment.

The original RHRS pipe stress analysis was performed utilizing the PIPESTRESS System 2010 computer program. A manual computation methodology was used to evaluate pipe stresses and pipe supports against the design criteria, considering SGR/Uprate changes in pressure, temperature and flow parameters. Additional evaluations are not required for instances resulting in load changes which are ≤ 5% of the original load, since the impact of 5% load increases are considered insignificant. The load increases were less than 5% and the supports were accepted without further review.

The results of the evaluation of the RHRS, and its individual components, to satisfy applicable design and licensing bases in accordance with the FSAR under SGR/Uprate conditions are presented in Section 2.11.4. Acceptance criteria, relevant to the RHRS, are identified in Section 2.11.3.

2.11.3 Acceptance Criteria

The licensing bases for the RHRS are described in the FSAR. The RHRS function is to transfer decay heat of fission products and other residual heat from the reactor core at a rate such that specified acceptable design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

In addition, Regulatory Guides 1.1, 1.26, 1.29 and 1.139 are used as guidance (refer to FSAR Section 1.8 for CP&L's compliance to each RG).

The design, operation, and functional capabilities of the RHRS described in the FSAR, and as affected by the SGR/Uprate, were evaluated against the acceptance criteria. The results of these evaluations are described in Section 2.11.4.

2.11.4 Results

The SGR/Uprate does not change the RHRS, and its intended design functions, as described in the FSAR. There are no configuration changes required for the RHRS under SGR/Uprate conditions and any changes to process conditions (e.g., an increased decay heat load) are within the capabilities of the existing design.

The RHR system is being evaluated by calculation with respect to overpressure protection. Based on the results of the calculation the RHR suction relief valves will be modified, if necessary.

2.11.5 Conclusions

The existing RHRS design is adequate to support SGR/Uprate. There are no required configuration changes to the system design. The existing system components are adequate to meet the SGR/Uprate conditions. RHRS heat removal functions and capabilities are not adversely affected by SGR/Uprate and remain in accordance with the existing FSAR.

The results obtained with the Delta 75 RSG at the uprated NSSS power level of 2912.4 MWt bound operation of the Delta 75 RSG at the current NSSS power level of 2787.4 MWt.

2.11.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.4.9.1 RCS Pressure/Temperature Limits
   3/4.5.2 ECCS Subsystems - T_{avg} Greater than or Equal to 350°F
   3/4.5.3 ECCS Subsystems - T_{avg} Less than 350°F
   3/4.5.4 Refueling Water Storage Tank
   3/4.6.3 Containment Isolation Valves
   3/4.9.8 Residual Heat Removal and Coolant Circulation

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"

9. RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," (Rev. 0, 11/70)

10. RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste Containing Components of Nuclear Power Plants," (Rev. 3, 02/76)

11. RG 1.29, "Seismic Design Classification," (Rev. 3, 09/78)

12. RG 1.82, "Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident," (Rev. 2, 05/96)

13. RG 1.141, "Containment Isolation Provisions for Fluid Systems," (Rev. 0, 04/78)

14. RG 1.139, "Guidance for Residual Heat Removal (for Comment)," (05/1978)

15. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants"
    Criterion 2, "Design Bases for Protection Against Natural Phenomena"
    Criterion 4, "Environmental and Dynamic Effects Design Bases"
    Criterion 19, "Control Room"
    Criterion 34, "Residual Heat Removal"

16. Westinghouse Calculation CS-FSE-99-136

2.12 Safety Injection System

The Harris Nuclear Plant (HNP) Safety Injection System (SIS) has been evaluated to determine its performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.12.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the function of the SIS is to deliver borated water directly to the Reactor Coolant System (RCS) subsequent to a loss of coolant accident (LOCA), main steam line break (MSLB) or any other event which reduces the RCS inventory.

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not alter existing SIS design functions. The SIS remains capable of satisfying regulatory commitments in accordance with the existing FSAR.

Configuration Changes:

There are no configuration changes associated with the SIS under the SGR/Uprate conditions.

Revised Process Conditions:

Under SGR/Uprate, containment pressure, containment sump temperature and containment sump water level are increased following the most limiting design bases accidents. These changes to process conditions are minor and within the capabilities of the existing SIS design.

2.12.2 Description of Analyses and Evaluations

The SIS was evaluated to ensure that, following the SGR/Uprate, the system remains capable of performing required functions in accordance with the existing licensing bases specified in the FSAR.

The post LOCA Long Term Core Cooling analysis was performed to demonstrate, that under SGR/Uprate conditions, the continued cooling of the core can be maintained for the long term. The analysis is based on the SIS having a minimum SI flow capacity, sufficient for both the cold leg and hot leg recirculation cooling mode, to prevent core re-uncovery and boron precipitation. The results of the evaluation have shown that the minimum SI flow is sufficient to perform the accident mitigating function.

RHR/SI check valves to the cold and hot legs were reviewed to ensure that valve performance was not adversely affected due to SGR/Uprate. The results indicate that these valves will perform as required under SGR/Uprate conditions.

The changed process conditions that would potentially affect pump flow rate are a higher containment sump temperature and a small increase in containment sump water level. The SGR/Uprate does not adversely affect the flow rates of the charging pumps during recirculation mode; i.e., runout conditions are not exceeded. Therefore, the existing charging pumps are adequate for the intended function.

The change in system pressures, temperatures and flow due to SGR/Uprate are small. Existing instruments have sufficient range such that instrumentation and control functions are not affected.

The original SIS pipe stress analysis was performed utilizing the PIPESTRESS System 2010 computer program. The manual computation methodology was used to evaluate pipe stresses and pipe supports against the design criteria. This methodology is based on performing the manual computations using existing results, considering SGR/Uprate changes in pressure, temperature and flow parameters. Additional evaluations are not required for instances resulting in load changes which are ≤ 5% of the original load, since the impact of 5% load increases are considered insignificant. The supports were accepted without further review.

The results of the evaluation of the SIS, and its individual components, to satisfy applicable design and licensing bases in accordance with the FSAR are presented in Section 2.12.4. Acceptance criteria, relevant to the SIS, are identified in Section 2.12.3.

2.12.3 Acceptance Criteria

The design, operation, and functional capabilities of the SIS described in the FSAR, and as affected by the SGR/Uprate, were evaluated against the acceptance criteria (capable of supplying flow rates specified in accident analyses without violating minimum/maximum assumptions). The results of these evaluations are described in various sections of the licensing report.

2.12.4 Results

The SGR/Uprate does not change the SIS, or its intended design functions, as described in the FSAR. The current design of the SIS bounds the SGR/Uprate conditions and the individual components remain capable of performing their functions without being adversely impacted by the SGR/Uprate.

The piping system and valve components are not impacted by the SGR/Uprate since the flow rates and velocities are not increased. The existing piping, valves and other system piping components are not required to be modified to accommodate the SGR/Uprate.

Existing instruments have sufficient range such that instrumentation and control functions are not affected. No instrumentation or control changes are required for the SIS as a result of SGR/Uprate.

SGR/Uprate does not adversely affect the ability of the SIS to continue to satisfy required functions, including removal of residual and decay heat of fission products from the reactor.

2.12.5 Conclusions

The existing SIS design is adequate to support SGR/Uprate. There are no required configuration changes to the system design. The existing system components are adequate to meet the SGR/Uprate conditions. SIS heat removal functions and capabilities are not adversely affected and remain in accordance with the existing FSAR.

The results obtained with the Delta 75 RSG at the uprated NSSS power level of 2912.4 MWt bound operation of the Delta 75 RSG at the current NSSS power level of 2787.4 MWt.

2.12.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.5 Emergency Core Cooling System
   3/4.6.3 Containment Isolation Valves

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"

9. RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," (Rev. 0, 11/70)

10. RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste Containing Components of Nuclear Power Plants," (Rev. 3, 02/76)

11. RG 1.29, "Seismic Design Classification," (Rev. 3, 09/78)

12. RG 1.82, "Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident," (Rev. 2, 05/96)

13. RG 1.141, "Containment Isolation Provisions for Fluid Systems," (Rev. 0, 04/78)

14. GL 89-10 "Safety Related Motor-Operated Valve Testing and Surveillance" (Rev. 0, 06/28/89)

15. 10CFR50.46, Acceptance Criteria for ECCS for Light Water Nuclear Power Reactor

16. 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants"
Criterion 2, "Design Bases for Protection Against Natural Phenomena"
Criterion 4, "Environmental and Dynamic Effects Design"
Criterion 17, "Electric Power Systems"
Criterion 27, "Combined Reactivity Control System Capability"
Criterion 35, "Emergency Core Cooling"

17. 10CFR50, Appendix K, ECCS Evaluations Models

2.13 Containment Spray System

The Harris Nuclear Plant (HNP) Containment Spray System (CSS) has been evaluated to determine its performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.13.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the function of the CSS is to remove the heat and fission products that may be released into the containment atmosphere following a loss-of-coolant accident (LOCA) or secondary system pipe rupture by spraying borated sodium hydroxide solution (NaOH) into the Containment. The system removes heat under post-accident steam-air mixture conditions for pipe breaks, up to and including a double-ended break of the largest reactor coolant pipe or a double ended break of the largest main steam line inside Containment. The system consists of two redundant trains each containing a spray pump, piping, valves, spray headers and spray nozzles. The fission product removal function of the CSS is carried out in conjunction with its heat removal function. The CSS removes radionuclides from the containment atmosphere following a LOCA by adding controlled amounts of NaOH to the containment spray water. The CSS is designed to function during accident conditions. The CSS is also designed to operate in support of periodic testing.

The system has two principal modes of operation. During injection mode, the CSS uses borated water supplied from the Refueling Water Storage Tank (RWST). During long-term recirculation mode, the system utilizes borated water from the containment sump. The recirculation mode is initiated when a low-low level is reached in the RWST. During this mode, the pump suction is transferred from the RWST to the containment sump by opening the recirculation line valves and closing the RWST suction valves to the containment spray pumps. This switchover is accomplished automatically.

The absorption of elemental iodine is achieved by maintaining the pH in the initial spray stream (injection phase) between a value of not less than 7.0 and not more than 11.0. The introduction of NaOH into the containment sprays, in addition to the elemental iodine removal feature, also minimizes the effects of stress corrosion cracking on stainless steel mechanical systems and components. The quantity of chemical added is sufficient to achieve a sump solution pH above 7.0 at the onset of the recirculation phase and 8.5 by the end of sodium hydroxide addition.

Consistent with the existing design and in accordance with the FSAR, the minimum inventory of the RWST (i.e., 436,000 gallons) is used to determine the longest duration the CSS will be in the injection mode prior to switching to the recirculation mode (containment sump suction). Based on the minimum RWST volume, as specified by the Technical Specifications, the RWST low-low level alarm setpoint will be reached in approximately 36 minutes. This time was then used in revised LOCA analyses as discussed in the BOP LR Section 2.24. The minimum RWST volume is also used to determine the flood level in the containment sump.

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change existing CSS design functions.

Configuration Changes:

There are no configuration changes for the CSS under the SGR/Uprate conditions. There is a change required in the volume of NaOH in the Spray Additive Tanks (SAT); see Section 2.13.2.

Revised Process Conditions:

Under the SGR/Uprate conditions, the pressures and temperatures inside containment have increased, following a postulated loss of coolant accident (LOCA) or main steam line break (MSLB).

2.13.2 Description of Analyses and Evaluations

The CSS was evaluated to ensure that, following the SGR/Uprate, the system remains capable of performing required functions in accordance with the FSAR.

The required RWST inventory is unchanged and sufficient to enable the CSS to meet its containment heat removal function and control containment pressure transients.

Conditions inside Containment following a design basis accident were analyzed assuming that the containment spray train with the lowest spray flow capacity is operable since this results in the longest fill-up time duration. The new Containment Analyses for the SGR/Uprate are described in BOP LR Section 2.24. The resulting spray effectiveness coeffiecient is discussed in BOP LR Section 2.22.

Following the SGR/Uprate, the calculated minimum CSS flow has been calculated at the containment design pressure and the assumed spray flow reduced from 1832 gpm to 1730 gpm.

The existing Containment Spray Pump characteristics [i.e., flow rate, dynamic head, Brake Horsepower (BHP), Net Positive Suction Head Available (NPSHa), and Net Positive Suction Head Required (NPSHr), etc.] are not impacted as a result of SGR/Uprate.

Due to the larger primary side on the RSGs, the RCS volume, used in the pH analysis, has increased. Revised calculations were performed to evaluate the sump and spray pH against the required limits.

The results of the evaluation of the CSS and its individual components to satisfy applicable design and licensing bases in accordance with the FSAR are presented in Section 2.13.4. Acceptance criteria, relevant to the CSS, are identified in Section 2.13.3.

2.13.3 Acceptance Criteria

The licensing basis for the CSS and its individual components is described in the FSAR. CSS is designed to remove heat and fission products that may be released into the containment atmosphere following a LOCA or secondary system pipe rupture by spraying borated sodium hydroxide solution into the containment.

Acceptance criteria specifically relevant to SGR/Uprate include:

1. The containment spray performance as described in Section 2.13.2 is provided as input to the containment analysis described in BOP LR Section 2.24.

2. The increase in containment temperature does not adversely upset the integrity of the containment spray system piping and supports.

3. The addition of NaOH to the containment spray meets the pH requirements listed in Section 2.13.1.

4. The Post-LOCA water level provides adequate NPSH for the Containment Spray Pumps.

5. The performance of the containment spray system is input to the analysis of the removal of fission products from the containment atmosphere described in BOP LR Section 2.22.

The design, operation, and functional capabilities of the CSS described in the FSAR, and as affected by the SGR/Uprate, were evaluated against the acceptance criteria. The results of these evaluations are described in Section 2.13.4.

2.13.4 Results

The SGR/Uprate does not change the CSS and its functions described in the FSAR. As described in Section 2.13.1, the CSS does not require configuration changes for the SGR/Uprate. Revised process conditions are within the existing design bases of the system and its components. The CSS meets all the functional requirements at SGR/Uprate conditions. Therefore, the existing CSS design is acceptable for the SGR/Uprate and no system design modifications are required.

The change in the minimum pH value is negligible.

For the maximum pH case, the evaluation of the required change is not complete. As indicated in the transmittal letter for these licensing reports (HNP-00-142), an additional Technical Specification change (3/4.6.2.2.a) regarding spray additive tank limits is anticipated subsequent to completion of the evaluation.

As discussed in Section 2.13.2 adequate NPSH is available for the containment spray pumps.

The existing CSS and its components are adequately designed to support the functional requirements to mitigate containment transients. The CSS evaluation supports that the existing design meets all system functional requirements at SGR/Uprate conditions.

2.13.5 Conclusions

No configuration changes are required to the system design as a result of the SGR/Uprate. The existing system components remain adequate to meet the SGR/Uprate conditions.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.13.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.5.4 Refueling Water Storage Tank
   3/4.6.2.1 Containment Spray System Limiting Condition for Operation/Surveillance Requirements
   3/4.6.2.2 Spray Additive System Limiting Condition for Operation/Surveillance Requirements

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2)," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

2.14 Steam Generator Blowdown System

The Harris Nuclear Plant (HNP) Steam Generator Blowdown System (SGBS) has been evaluated to determine its performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.14.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR) Section 10.4.8, the Steam Generator Blowdown System (SGBS) is used in conjunction with the Secondary Sampling System to control the chemical composition of water in the secondary side of the steam generator shells within specified limits and to prevent the buildup of corrosion products. The Steam Generator Blowdown System removes contaminants and corrosion product accumulations from the steam generators to maintain secondary water chemistry within prescribed limits.

Configuration Changes:

The RSGs are provided with two blowdown connections, a tube sheet connection and a shell connection. The installation of the RSGs includes a piping modification that relocates system valves from inside to outside the secondary shield wall.

Revised Process Conditions:

SGBS continuous allowable flow rates will increase slightly along with a reduction in the maximum allowable SGBS flow per steam generator as a result of SGR/Uprate.

2.14.2 Description of Analyses and Evaluations

The SGBS was evaluated to ensure that, following the SGR/Uprate, the system remains capable of performing its required functions.

The SGBS and its components were evaluated by comparing the component capacities and design capabilities against the changes expected in the SGR/Uprate operating conditions. This evaluation considered the following items with respect to design and operational requirements to support the SGR/Uprate:

- Pressure, temperature, and flow
- Pipe size, erosion-corrosion rate, and code compliance
- Pipe Rupture/Jet Impingement
- Valves/Components Capabilities
- Water Hammer

2.14.3 Acceptance Criteria

The acceptance criteria for the system is to maintain the design and licensing bases for the SGBS as it is described in the FSAR. A reduction in the maximum blowdown rate as identified in section 10.4.8.1,item 2 in the FSAR is required for the SGR/Uprate. This reduction is from a current value of 3.5%(~300 gpm/SG) of the current maximum steaming rate, to ~1.7% (~200 gpm/SG) of the SGR/Uprate steaming rate. This reduction ensures that the maximum SGBS flow rate (currently under administrative control) is within requirements of the continuous calorimetric program when the plant is operating above 35% power.

2.14.4 Results

The SGR/Uprate does not change the system and component functions as described in the FSAR. Although the SGR/Uprate involves slightly different process flow rate limitations, the functional capabilities of the system will be maintained.

The SGBS components were evaluated to determine their compatibility with the revised process flow conditions and were found to be acceptable since the original design capacities of these components bound the SGR/Uprate conditions.

The SGBS maximum operating pressure and temperature will remain bounded by the existing design.

The Technical Specifications Section 3.6.3.1 has a limiting condition of operation (LCO) for the operability of the SGBS containment isolation valves. However, the operation of the SGBS under SGR/Uprate does not alter the limiting condition of operation or the Tech Spec surveillance requirements of the containment isolation valves. Hence, there is no impact on the Technical Specifications requirements.

2.14.5 Conclusions

The revised SGBS process conditions under SGR/Uprate are bounded by the existing design and remain within the SGBS design limits.

The SGR/Uprate will not impact the SGBS performance since the increase in continuous allowable normal blowdown flow rate is negligible and remains bounded by the maximum allowable flow rate. The SGR/Uprate will not increase the rate of addition of dissolved solids or particulates into the steam generators. The changes in SGBS process conditions, pipe routings and valve locations are acceptable in terms of system hydraulics.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.14.6 References

1. HNP Final Safety Analysis Report
2. HNP Technical Specifications, Section 3.6.3.1

2.15 Heating Ventilation and Air Conditioning (HVAC)

2.15.1 Containment Ventilation and Cooling Systems

The Harris Nuclear Plant (HNP) Containment Ventilation and Cooling Systems have been evaluated to determine their performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.15.1.1 Introduction and Background

The Containment Ventilation and Cooling Systems evaluated in this BOP Licensing Report (LR) section include the following:

- Containment Cooling System,
- Elevator Machine Room Cooling System,
- Primary Shield Cooling System,
- Reactor Supports Cooling System,
- Vacuum Relief System,
- Normal Purge System,
- Pre-Entry Purge System,
- Hydrogen Purge System, and
- Airborne Radioactivity Removal System.

The Containment Spray System is evaluated in BOP LR Section 2.13.

As described in the Final Safety Analysis Report (FSAR), the Containment Cooling System is designed to provide cooling and maintain the average temperature of the containment atmosphere during normal operations. The system consists of four safety related fan cooler units (two trains of two each) and three non-safety fan coil units. During normal operation the fan coil units and fan coolers continuously remove the heat dissipated inside the containment. During a Design Basis Accident (DBA) the containment fan coolers, in conjunction with the Containment Spray System, provide heat removal capability and limit the transient temperature and pressure inside containment.

The Elevator Machine Room Cooling System provides cooling and ventilation to the elevator machine room during normal operation. This system provides no post-accident function.

The Primary Shield Cooling System and the Reactor Supports Cooling System provide cooling air to the reactor vessel primary shield cavity and the reactor support/concrete interface. The systems function during normal operations and are not required to function during accident conditions.

The Vacuum Relief System maintains the structural integrity of the Containment Building during an inadvertent actuation of the containment spray system by providing vacuum relief to limit the differential pressure across the containment within design limits.

The Normal Purge System provides pressure relief via a low volume purge. The system also functions to create a slight vacuum inside containment to prevent outleakage during normal operation.

The Pre-Entry Purge System reduces the concentration of radioactivity in the containment atmosphere to an acceptable level by purging the containment with a high volume of outside air to control airborne radioactivity and reduce personnel occupational exposure.

The Hydrogen Purge System provides capability of hydrogen purge, as a backup to hydrogen recombiners, following a LOCA and after the containment pressure is reduced to atmospheric pressure.

The Airborne Radioactivity Removal System removes airborne particulate radioactivity and reduces the concentration of radioactive iodine in the containment atmosphere by recirculating the air through HEPA filters and charcoal adsorbers during normal operation only.

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change Containment Ventilation and Cooling Systems design functions.

Configuration Changes:

There are no configuration changes to the Containment Ventilation and Cooling Systems under the SGR/Uprate conditions.

2.15.1.2 Description of Analyses and Evaluations

The Containment Ventilation and Cooling Systems were evaluated to ensure that, following the SGR/Uprate, the systems remain capable of performing required design functions in accordance with existing licensing bases specified in the FSAR.

The results of the evaluation of the Containment Ventilation and Cooling Systems and their individual components to satisfy applicable design and licensing bases in accordance with the FSAR are presented in Section 2.15.1.4. Acceptance criteria, relevant to the Containment Ventilation and Cooling Systems, are identified in Section 2.15.1.3.

2.15.1.3 Acceptance Criteria

The licensing basis for the Containment Ventilation and Cooling Systems are described in the FSAR. The following acceptance criteria are relevant to those SGR/Uprate activities that involve the Containment Ventilation and Cooling Systems:

1. For normal operation, the Containment Cooling System is required to maintain average indicated containment air temperature below the value (120°F).

2. For accident conditions, the Containment Cooling System is required to remove energy from the containment atmosphere in order to maintain the containment within the containment design limits.

3. The Vacuum Relief System maintains the structural integrity of the Containment Building by providing vacuum relief to prevent differential pressure between the containment and the outside atmosphere from exceeding the design value of negative 2.0 psid.

4. The Hydrogen Purge System provides capability of hydrogen purge, as a backup to hydrogen recombiners, following a LOCA to ensure the hydrogen levels are maintained below four percent by volume in the Containment.

The Control Rod Drive Mechanism Cooling System, Normal and Pre-Entry Purge System, Airborne Radioactivity Removal System, Elevator Machine Room Cooling System, Primary Shield Cooling System and Reactor Supports Cooling System are not impacted by the power uprate conditions.

The design, operation, and functional capabilities of the Containment Ventilation and Cooling Systems described in the FSAR, and as affected by the SGR/Uprate, were evaluated against the acceptance criteria. The results of these evaluations are described in Section 2.15.1.4.

2.15.1.4 Results

The containment cooling system performance for normal operation was evaluated with SGR/Uprated conditions. Based on the engineering evaluations presented in the evaluation report for Containment Ventilation and Cooling System, including fans, cooling coils, and other components, the following conclusions are provided:

- There is no impact on the Containment Ventilation and Cooling Systems due to the changing operating conditions of the power uprate program.

- The Containment Ventilation and Cooling System has adequate cooling capacity to maintain the space design conditions with a service water flow rate of as low as 1300 gpm at 95° F to each of the safety-related containment fan coolers.

2.15.1.5 Conclusions

The evaluation shows that the SGR/Uprate does not impact the existing design of the Containment Ventilation and Cooling Systems. The Containment Ventilation and Cooling System has adequate cooling capacity to maintain the space design conditions.

The results of evaluations of the Containment Ventilation and Cooling Systems assuming the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.15.1.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications

3/4.6.1.4	Internal Pressure
3/4.6.1.5	Air Temperature
3/4.6.1.7	Containment Ventilation System
3/4 6.2.3	Containment Cooling System
3/4.6.3	Containment Isolation Valves
3/4.6.5	Vacuum Relief System
5.2	Containment

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2 ,"Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

2.15.2 Reactor Auxiliary Building HVAC System

The Harris Nuclear Plant Reactor Auxiliary Building (RAB) Heating, Ventilation, and Air Conditioning (HVAC) System has been evaluated to determine its performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

The evaluation shows that the SGR/Uprate does not impact the existing design of the RAB HVAC System. There are no configuration changes or process condition changes to the RAB HVAC System per SGR/Uprate. Consequently, the RAB HVAC System will continue to satisfy the existing requirements in accordance with the FSAR.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.15.3 Fuel Handling Building HVAC System

The Harris Nuclear Plant (HNP) Fuel Handling Building (FHB) Heating Ventilation and Air Conditioning (HVAC) has been evaluated to determine its performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

The evaluations show that the SGR/Uprate alone does not adversely affect the existing design of the FHB HVAC System.

The results obtained with the Model Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Model Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.15.4 Essential and Non-Essential Services Chilled Water Systems

The Harris Nuclear Plant Essential and Non-Essential Services Chilled Water Systems have been evaluated to determine their performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt. This evaluation takes into account the bounding heat loads in Spent Fuel Pools A/B and activation of Spent Fuel Pools C/D with a heat load of 1 MBTU/Hour. The combination of effects on chilled water are referred to as the (proposed change).

2.15.4.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the safety-related Essential Services Chilled Water System consists of two independent trains of equipment and piping of 100% capacity each (one operating and one standby). Each train is a closed loop system consisting of a packaged water chiller, an expansion tank, a makeup water system, a chemical addition tank, a chilled water pump and distribution piping. The condenser water for the chillers is supplied from the safety-related Service Water System.

The Essential Services Chilled Water System provides chilled water to the cooling coils of the air handling units in the:

- Control Room Air Conditioning System,
- Reactor Auxiliary Building (RAB) Engineered Safety Feature Equipment Cooling System,
- RAB Non-Nuclear Safety Ventilation System,
- RAB Switchgear Rooms Ventilation System,
- RAB Electrical Equipment Protection Rooms Ventilation System, and
- Fuel Handling Building (FHB) Spent Fuel Pool Pump Room Ventilation System.

The Essential Services Chilled Water System is designed to provide chilled water at a nominal 44°F to the cooling coils served, and automatic isolation of the non-safety related RAB Ventilation System following a Safety Injection Actuation Signal (SIAS).

The non-safety related Non-Essential Services Chilled Water System is a closed loop system consisting of two 50% capacity chillers in series, an expansion tank, a makeup water system, a chemical addition tank, two 100% chilled water pumps (one operating and one standby) and distribution piping. The condenser water for the chillers is supplied from the non-safety related Service Water System.

The Non-Essential Services Chilled Water System provides chilled water to the cooling coils of the air handling units in the FHB Spent Fuel Pools and Operating Floor Air Conditioning System, and the Waste Processing Building (WPB) Heating Ventilation and Air Conditioning (HVAC) System. The Non-Essential Services Chilled Water System is designed to provide chilled water at a nominal 44°F to the cooling coils served.

The Essential and Non-Essential Services Chilled Water Systems have been evaluated considering the combined effects from the proposed change. The combined impact of implementing these projects bounds the impact of implementing one or more of these projects. The impact of the proposed change does not change existing Essential and Non-Essential Services Chilled Water Systems design functions. The existing Essential and Non-Essential Services Chilled Water Systems remain capable of satisfying regulatory commitments in accordance with the design bases, described in the FSAR.

Configuration Changes:

There are no configuration changes required for the Essential and Non-Essential Services Chilled Water Systems under the proposed change conditions.

Revised Process Conditions:

There are no revised process conditions to the Essential and Non-Essential Services Chilled Water Systems, following the proposed change.

2.15.4.2 Description of Analyses and Evaluations

The Essential and Non-Essential Services Chilled Water Systems were evaluated to ensure that, following the proposed change, the chilled water systems remain capable of performing required functions in accordance with existing licensing bases specified in the FSAR.

The capabilities of the Essential and Non-Essential Services Chilled Water Systems, including their components/equipment and the existing design bases were first identified by reviewing available documentation. Then the impact on the system and plant equipment, as a result of the proposed change was evaluated.

The evaluation determined that there is minimal impact on the Essential and Non-Essential Services Chilled Water Systems due to the changing operating conditions of the proposed change. Consequently, the Essential and Non-Essential Services Chilled Water Systems remain capable of satisfying their intended functions.

The results of the evaluation of the Essential and Non-Essential Services Chilled Water Systems and their individual components to satisfy applicable design and licensing bases in accordance with the FSAR are presented in Section 2.15.4.4.

2.15.4.3 Acceptance Criteria

The licensing bases for the Essential and Non-Essential Services Chilled Water Systems are described in the FSAR. The Essential and Non-Essential Services Chilled Water Systems must provide sufficient heat removal capability to the associated HVAC Systems to ensure that area temperatures are maintained within design limits.

Regulatory Guide 1.29, "Seismic Design Classification" is used as guidance (refer to FSAR Section 1.8 for CP&L's compliance to the Regulatory Guide).

The design, operation, and functional capabilities of the Essential and Non-Essential Services Chilled Water Systems described in the FSAR, and as affected by the proposed change, were evaluated against the acceptance criteria.

2.15.4.4 Results

The proposed change does not change Essential and Non-Essential Services Chilled Water Systems and components functions, as described in the FSAR. As stated in Section 2.15.4.1, the proposed change involves no configuration changes and no process condition changes to the Essential and Non-Essential Services Chilled Water Systems. Area temperatures profiles will be maintained in accordance with existing design bases requirements, following proposed change.

The components of the Essential and Non-Essential Services Chilled Water Systems were evaluated to determine their capability, following the proposed change. The water chillers, pumps, tanks and piping are adequate for the proposed conditions, since the proposed change cooling loads, process conditions and chemistry are consistent with the existing design.

2.15.4.5 Conclusions

The results of the evaluations in Section 2.15.4.4 show that the existing design of the Essential and Non-Essential Services Chilled Water Systems is not impacted by the proposed change.

Cooling load demands, following the proposed change are consistent with the capabilities of the existing Essential and Non-Essential Services Chilled Water Systems. The chilled water systems will continue to circulate chilled water at the required temperature to the cooling coils of the air handling units serving their respective HVAC systems. This will ensure the transfer of thermal loads generated in the various areas of the plant to the Service Water System. Consequently, the Essential and Non-Essential Services Chilled Water Systems will continue to satisfy the existing requirements in accordance with the FSAR.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.15.4.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.7.13 Essential Services Chilled Water System

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"

9. RG 1.29, "Seismic Design Classification," (Rev. 3, 9/78)

10. 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants"
    Criterion 2, "Design Bases for Protection Against Natural Phenomena"
    Criterion 4, "Environmental and Dynamic Effects Design Bases"
    Criterion 44, "Cooling Water"
    Criterion 45, "Inspection of Cooling Water"
    Criterion 46, "Testing of Cooling Water"

2.16 Structural Analyses

2.16.1 Reactor Building - Structural Analysis

A structural analysis of the Harris Nuclear Plant (HNP) Reactor Building (RB) has been conducted to ensure the structural adequacy of the building and its structural components. This analysis reflects operation of the plant with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.16.1.1 Introduction and Background

The RB consists of a steel-lined, reinforced concrete containment structure, the internal reinforced concrete structures and the internal steel structures. The containment structure encloses the internal concrete and steel structures.

As described in the Final Safety Analysis Report (FSAR), the function of the RB is to house the reactor pressure vessel, pressurizer, steam generators, reactor coolant pumps and piping, and portions of the engineered safety features systems. The containment structure protects the Reactor Coolant System (RCS), and other systems and components housed within the RB, from site environmental conditions including earthquake, tornado, and external missile loading conditions. The Containment is designated as Safety Class 2, Seismic Category I. The Containment is designed to withstand the consequences of postulated design basis accidents (DBAs) (e.g., Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), and Feedwater Line Break (FWLB)) in accordance with the FSAR. The Containment limits the release of radioactive fission products to the environment during and after postulated accidents. It also provides biological shielding during normal and post-accident operations.

The internal concrete and steel structures are designed to Seismic Category I requirements. The internal structures provide support functions for the NSSS equipment during all operational phases. In the unlikely event of an accident, the Containment and its internal structures serve to mitigate the consequences of the accident by protecting safety-related equipment and functions from the effects of the accident.

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change existing RB design functions.

Configuration Changes:
There are no configuration changes to the RB structure under SGR/Uprate conditions.

Revised Process Conditions:
Any changes to structural impact due to revised process conditions can be accommodated by the existing structural design margin. The revised plant operating conditions, which potentially affect the RB, are due to increase in pressure, temperature and pipe support and pipe break loading.

2.16.1.2 Description of Analyses and Evaluations

The RB was evaluated to ensure that, following SGR/Uprate, the structure remains capable of performing required functions in accordance with existing licensing bases specified in the FSAR. The following design parameters, which potentially affect the structural adequacy of the RB, were evaluated:

- The Containment accident pressure and temperature transients, and the maximum pressure and temperature experienced during the DBA events.

- The differential pressures across the structural components and the intensity of jet impingement loadings on the localized surfaces and the RCS equipment support loads, as well as pipe break and pipe support loadings.

- Increased accident temperature loadings, increased pipe break loadings and increased pipe support loadings that have an affect on the RB steel internal structures.

1. <u>Concrete Containment and Steel Liner</u>

The Containment structure was analyzed for the impact of the maximum pressure and temperature during the DBA events. Based on the analysis, the maximum pressure inside the containment is 56.5 psia or 41.8 psig, which is well within the maximum 45 psig design pressure used in the original concrete containment design. The SGR/Uprated accident liner temperature is 255.3°F, which is smaller than the liner temperature of 294°F used in the actual design of the concrete containment. Therefore, the SGR/Uprate has no effect on the structural adequacy of the concrete Containment Structure.

The accident temperature used in the design of liner and studs varies from 248°F to 262°F. Analysis has shown that the liner and studs are structurally adequate for an accident temperature of 262°F. Therefore, they are acceptable for the revised liner accident temperature with the considerations of SGR/Uprate.

2. <u>Personnel Airlocks, Equipment Hatch, and Other Penetrations</u>

For personnel airlocks and the equipment hatch, the original maximum design temperatures of 248°F for portions in contact with concrete and 263°F for portions without the concrete backing were based on the containment liner and internal steel accident temperatures, respectively. One side of the personnel airlocks and the equipment hatch is directly exposed to the Reactor Auxiliary Building (RAB) or the Fuel Handling Building (FHB). These buildings have a maximum normal operating temperature of 104°F in accordance with FSAR Section 3.8.4.3.1. Since heat will be directly transferred through the penetrations to the RAB or FHB, the accident temperature at personnel airlocks and the equipment hatch would be much less than those of the liner and internal steel structures. The revised accident temperatures for liner and internal steel structures are determined to be 255.3°F for potions in contact with concrete and 264.3°F for portions not in contact with concrete. The small temperature increase will not raise a concern since they can be easily compensated by heat loss to the RAB or FHB through the penetrations. The SGR/Uprate does not adversely affect these other penetrations. Therefore, the design of containment penetrations including personnel locks and the equipment hatch is adequate for the SGR/Uprate.

3. Internal Reinforced Concrete Structures

The design loads for the containment internal concrete structures, which may change with the SGR/Uprate conditions, are the compartment differential pressure, pipe rupture and pipe support loads, RCS equipment support loads and the accident temperature.

The internal reinforced concrete structures of the containment have been evaluated for any affect on walls and floors following compartment pressurization. The differential pressures across the primary and secondary shield walls with the SGR/Uprate consideration remains unchanged. Therefore, they are structurally adequate for the SGR/Uprate.

The differential pressure loads for compartments do not exceed the original design values with respect to the SGR/Uprate. The accident temperature for the concrete mat is approximately equal to the maximum operating temperature as used in the original design because the duration of the increased accident temperature is very short. Therefore, changes to accident temperature loads following SGR/Uprate have not adversely affected the original mat design.

Some of the RCS equipment support loads have increased, but the increases are minimal. In addition, the effects of the RCS support loads on the internal concrete structures are primarily at the localized support areas. The overall effects of these minimal equipment support load increases on the global internal concrete structures are negligibly small. The effects of pipe support load changes on internal reinforced concrete structures are minor. It was demonstrated that the original design margins of internal reinforced concrete structures could accommodate the pipe support load changes. The accident temperature for the internal concrete structures is equal to the maximum operating temperature as used in the original design because the duration of the increased accident temperature is very short. Therefore, there is no impact on the design adequacy of the internal concrete structures following the SGR/Uprate.

The effects of pipe rupture and pipe support load changes on internal reinforced concrete structures are addressed in more detail in Item 5 below.

4. Internal Steel Structures

The internal steel structures are three main platforms in the annulus space between the secondary shield wall and the containment wall, various equipment and component supports, and miscellaneous steel structures. These steel structures were re-analyzed for the temperature loadings and pipe reaction loads resulting from the SGR/Uprate. They were designed for a maximum Design Basis Accident (DBA) temperature and pipe support loads, as applicable. The containment design pressure has no impact on the internal steel structures. The majority of the steel structures are not affected by temperature loads because their connections have slotted holes to allow for expansion/contraction.

Some steel structures were evaluated for an accident temperature higher than or equal to 258°F during the original plant design. The base temperature used in the evaluation of these steel structures was 60°F. However, as shown in the analysis, the effective base temperature is higher than 60°F. Using the conservatively lower base temperature of 60°F will provide sufficient margin to accommodate the small increase in accident temperature. Therefore, the accident temperature following SGR/Uprate will have no impact on the structural adequacy of these steel structures.

Some steel structures were evaluated on a case by case basis. The evaluations indicated that the accident temperature load case is not the governing design load case for most of the structures evaluated. For the controlling load combination, which includes the accident temperature and pipe rupture loads, the maximum calculated stress interaction ratio is 0.924. This represents a minimum design margin of 7.6 % available to accommodate the increase of accident temperature. In addition, a base temperature of 60°F was conservatively used in the calculation of the thermal load for this critical case. Therefore, the accident temperature due to SGR/Uprate will have no impact on the structural adequacy of these steel structures.

With regard to the main steel platforms at EL. 236', 261' and 286', evaluations have been performed which accounted for a combination of concurrent loads such as dead loads, seismic loads, attachment loads and temperature loads. The forces/moments were reviewed during these evaluations for the highly stressed beams/connections and the review indicated that the governing load cases for the majority of critical steel members/connections do not include the temperature load. This is because these structures are generally free to expand under thermal loads due to the slotted holes provided in the connections and also because higher allowable stresses are allowed for cases which include the accident temperature. Therefore, there is sufficient design margin available to accommodate the potential increase of accident temperature in the main steel platforms due to the SGR/Uprate.

The effects of accident temperature on the Main Steam (MS) Towers and Main Steam/Feedwater (MS/FW) hard restraint structures were evaluated. The revised accident temperature is acceptable for these rupture restraint structures. (Reference 20)

Evaluations for SG access platforms were also performed. The revised accident temperature is acceptable to these SG access platform structures. (References 19 & 20)

The effects of pipe rupture and pipe support load changes on internal reinforced concrete structures are addressed in Item 5 below.

5. Pipe Support Loads

The pipe support loads for various fluid systems following SGR/Uprate were assessed. With the exception of the Class 1 Auxiliary Lines, there is no significant change in the pipe support loads following SGR/Uprate.

The increased new pipe reaction loads on the affected supports of Class 1 Auxiliary Lines were evaluated to assess their impact on the structural adequacy of the internal steel structures and internal concrete structure embedment anchorage. It was demonstrated that the original design margins are sufficient to accommodate the increased loads, and that the affected structural components can accommodate these loads without any compromise in the structural adequacy. Some of those pipe supports with load increases are attached to Reactor Building main steel platforms. The acceptability of the platform structures has been documented for the increase of pipe reaction loads.

Some internal steel and reinforced concrete structures were designed for the pipe rupture and jet impingement loads resulting from the original DBA conditions. The pipe rupture and jet impingement loads due to SGR/Uprate have not increased from their original design values. Therefore, there is no impact of pipe rupture and jet impingement on the structural adequacy of the internal steel and reinforced concrete structures following a SGR/Uprate.

The results of the evaluation of the RB and its individual components to satisfy applicable design and licensing bases in accordance with the FSAR are presented in Section 2.16.1.4. Acceptance criteria, relevant to the RB and the structural analysis, are identified in Section 2.16.1.3.

2.16.1.3 Acceptance Criteria

The licensing bases for Reactor Building is specified in FSAR Section 3.8.1.2. The design, operation, and functional capabilities of the Reactor Building structural analysis affected by the SGR/Uprate, were evaluated for the acceptance based on the licensing basis. The results of these evaluations are described in Section 2.16.1.4.

2.16.1.4 Results

The RB structural components, i.e., the Containment Structure, liner and penetrations, and reinforced concrete and structural steel internal structures, have been evaluated for their structural adequacy to function safely under the SGR/Uprate. It has been demonstrated that these structural components have more than adequate margins and require no physical "hardware" modifications.

Based upon the information as described above, the SGR/Uprate does not impact the structural adequacy of the RB. The structure remains capable of performing required functions following the SGR/Uprate.

2.16.1.5 Conclusions

The RB structural components, i.e., the Containment Structure, liner and penetrations, and reinforced concrete and structural steel internal structures, have been demonstrated to have more than adequate margins and require no physical "hardware" modifications.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.16.1.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.6.1 Primary Containment
   B3/4.6.1.4 Internal Pressure
   B3/4.6.1.6 Containment Structural Integrity
   5.2.2 Design pressure and Temperature
   6.8.4.K Containment Leakage Rate Testing Program

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1" dated October 1986

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"

9. RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste Containing Components of Nuclear Power Plants," (Rev. 3, 2/76)

10. RG 1.29, "Seismic Design Classification," (Rev. 3, 9/78)

11. RG 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," (Rev. 0, 6/73)

12. RG 1.59, "Design Basis Floods for Nuclear Power Plants," (Rev. 2, 8/77)

13. RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants" (Rev. 1, 12/73)

14. RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," (Rev. 0, 10/73)

15. RG 1.76, "Design Basis Tornado for Nuclear Power Plants," (Rev. 0, 4/74)

16. RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," (Rev. 1, 2/76)

17. RG 1.117, "Tornado Design Classification," (Rev. 1, 4/78)

18. 10CFR50, Appendix A, "General Design Criteria for the Nuclear Power Plants"
    Criterion 1, "Quality Standards and Records"
    Criterion 2, "Design Bases for Protection Against Natural Phenomena"
    Criterion 4, "Environmental and Dynamic Effects Design"
    Criterion 16, "Containment Design"
    Criterion 50, "Containment Design Basis"

19. ESR #97-00805, "Containment Mods"

20. ESR #97-00807, "Large Bore Piping Mods"

21. ESR #97-00808, "Small Bore Piping Mods"

22. ESR #97-00810, "SG Vessel and RCS Mods"

2.16.2 Reactor Auxiliary Building - Structural Analysis

A structural analysis of the Harris Nuclear Plant (HNP) Reactor Auxiliary Building (RAB) has been conducted to ensure the structural adequacy of the building and its structural components. The analysis reflects operation of the plant with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.16.2.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the function of the RAB is to house the engineered safeguards and supporting systems, switchgear room, sample rooms and the main Control Room. The RAB also houses the main steam tunnel. The RAB, a safety related structure, is designed to seismic Category I requirements and provides support and protection to plant personnel, structures, systems and components from radiation, postulated accidents and external hazards.

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change existing RAB design functions. The RAB remains capable of satisfying regulatory commitments in accordance with the existing FSAR following SGR/Uprate.

<u>Configuration Changes</u>:

There are no configuration changes to the RAB structure under SGR/Uprate conditions.

<u>Revised Process Conditions</u>:

Any changes to structural impact due to revised process conditions can be accommodated by the existing structural design margin. The revised plant operating conditions, which potentially affect the RAB, are due to increase in pressure, temperature and pipe rupture loads.

2.16.2.2 Description of Analyses and Evaluations

The RAB was evaluated to ensure that, following SGR/Uprate, the structure remains capable of performing its required functions in accordance with existing licensing bases specified in the FSAR. The following parameters with the potential to impact the structural adequacy of the RAB were evaluated:

- Changes in accident pressure and temperature loadings that are used for the design of the structural components of the Main Steam Tunnel and some of the internal structural components,

- Loadings due to the postulated pipe rupture of Feedwater, Auxiliary Feedwater and Steam Generator Blowdown Systems, including pipe break reaction loads and the jet impingement loads, and

- Localized effects of the reaction loads of the pipe supports and restraints on the structural components and embedments.

The Main Steam Tunnel reinforced concrete boundary was analyzed and designed for the accident pressurization effects to prevent over pressurization of the Main Steam Tunnel compartment. The high energy lines within the tunnel employ specialized "super pipe" design. As specified in FSAR Section 3.6, the steam tunnel is not subject to the pipe break or jet impingement loadings. Although "super pipe" construction is utilized for the high-energy lines, it was required that a non-mechanistic crack equivalent to the flow area of a single ended pipe break be considered in the Main Steam Tunnel area. The analysis considered the pressurization and environmental effects of the break.

The design peak accident pressure that was considered in the Main Steam Tunnel was 16 psig. The existing peak accident pressure is 6.47 psig; the re-evaluated peak accident pressure in the Main Steam Tunnel is 5.1 psig. The accident pressures in the remaining areas of the RAB have not changed as a result of the SGR/Uprate. Therefore, accident pressures under SGR/Uprate have no impact on the structural adequacy on the Main Steam Tunnel and the remaining RAB structures.

The platforms in the Main Steam Tunnel were re-evaluated for the accident temperature loadings resulting from the SGR/Uprate. These platforms are primarily supported off the massive pipe support structure steel members and the tunnel walls. These platforms were designed as Seismic Category I and are provided for proper access to the areas and equipment requiring monitoring and maintenance. These platforms were designed for various loads and their combinations including the temperature loadings resulting from the postulated accident events.

The increase in the maximum temperature in the Main Steam Tunnel will affect the platform steel. Global and localized deformations and localized yielding may occur as a result of this temperature increase but structural failure will not occur. There are no safety functions for these platforms. The SGR/Uprate does not introduce seismic II/I concerns with these platforms. Safety related power operated relief valve hydraulic actuator pressure indicators are attached to some of the platform framing members. These instruments are not required to be operational during and after the pipe break accident. Therefore, the elevated SGR/Uprate accident temperatures do not adversely affect the overall structural stability and adequacy of these platforms to perform required functions in accordance with the FSAR.

Loadings due to the postulated pipe rupture have been evaluated for the RAB high-energy pipe systems including the Feedwater System, Auxiliary Feedwater System, and SG Blowdown System. The results of these evaluations demonstrate that the original loadings bound any changes in these loadings due to the SGR/Uprate. Therefore, the postulated pipe rupture loads following SGR/Uprate have not adversely affected the structural adequacy of the RAB and its structural components.

In addition, localized effects of the reaction loads of the pipe supports and restraints on the anchorage embedments, pipe support frames or building framing steel have been shown to be insignificant due to SGR/Uprate. Therefore, the SGR/Uprate has no impact on the structural adequacy of the embedments and localized structural components of the RAB.

The results of the evaluation of the RAB and its individual components to satisfy applicable design and licensing bases in accordance with the FSAR are presented in Section 2.16.2.4.

2.16.2.3 Acceptance Criteria

The design, operation, and functional capabilities of the Reactor Auxiliary Building Structural Analysis described in the FSAR, and as affected by the SGR/Uprate, were evaluated for the acceptance. The results of these evaluations are described in Section 2.16.2.4.

2.16.2.4 Results

Based upon the information as described above, the SGR/Uprate does not impact the structural adequacy of the RAB. The structure remains capable of performing required functions in accordance with the FSAR, following SGR/Uprate.

The changes in accident pressures following SGR/Uprate have no impact on the structural adequacy of the Main Steam Tunnel and the remaining RAB structures. The elevated SGR/Uprate accident temperatures do not adversely affect the overall structural stability and adequacy of the platforms in the main steam tunnel to perform required functions in accordance with the FSAR. The postulated pipe rupture loads following SGR/Uprate do not adversely affect the structural adequacy of the RAB and its structural components. The SGR/Uprate has no impact on the structural adequacy of the embedments and localized structural components of the RAB.

In addition, the SGR/Uprate does not change or modify the RAB in any way that would diminish the protection by the existing design from external hazards. As a result, the RAB remains in accordance with the existing FSAR, which address environmental extremes and site proximity hazards. The SGR/Uprate also does not introduce any new external hazards, nor does it increase the severity or probability of existing external hazards including Turbine/Generator Missiles, as described in LR Section 2.4 "Turbine/Generator Evaluations."

There are no Technical Specifications pertaining to the RAB and therefore, the Technical Specifications are not impacted by the SGR/Uprate.

2.16.2.5 Conclusions

The SGR/Uprate does not adversely affect the RAB, which remains capable of performing its required functions in accordance with the existing FSAR. The revised peak accident pressure under SGR/Uprate conditions has no impact on the structural adequacy of the main steam tunnel. The increase in maximum temperature in the main steam tunnel under SGR/Uprate conditions will not affect the overall structural stability. The postulated pipe rupture loads following SGR/Uprate have not adversely affected the structural adequacy of the RAB and its structural components. The SGR/Uprate has no impact on the structural adequacy of the embedments and localized structural components of the RAB.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.16.2.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"

9. RG 1.29, "Seismic Design Classification," (Rev. 3, 9/78)

10. RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," (Rev. 1, 12/73)

11. RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," (Rev. 0, 10/73)

12. RG 1.76, "Design Basis Tornado for Nuclear Power Plants," (Rev. 0, 4/74)

13. RG 1.117, "Tornado Design Classification," (Rev. 1, 4/78)

14. 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plant"
    Criterion 1, "Quality Standards and Records"
    Criterion 2, "Design Bases for Protection Against Natural Phenomena"
    Criterion 4, "Environmental and Dynamic Effects Design Bases"

15. 10CFR50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"

2.16.3 Fuel Handling Building - Structural Analysis

A structural analysis of the Harris Nuclear Plant Fuel Handling Building (FHB) has been conducted to ensure the structural adequacy of the building and its structural components. This analysis reflects operation of the plant with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.16.3.1 Introduction and Background

The FHB is a reinforced concrete, seismic Category I structure, analyzed and designed for all applicable loads and their combinations, as described in Final Safety Analysis Report (FSAR) Section 3.8.4.3.

Configuration Changes:

There is no significant impact on the FHB and its internal structural components under the Steam Generator Replacement and Power Uprate Project (SGR/Uprate) conditions.

2.16.3.2 Results

The SGR/Uprate does not impact the structural adequacy of the FHB and its internal components.

2.16.3.3 Conclusions

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.16.3.4 References

1. HNP Final Safety Analysis Report

2. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

3. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

4. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

5. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

6. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

2.17 Electrical Reviews

2.17.1 Onsite Electrical Distribution and Main Power System

The Harris Nuclear Plant (HNP) Onsite Electrical Distribution Systems and Main Power System have been evaluated to determine their performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt. This licensing report section addresses the entire Onsite Electrical Distribution System and portions of the Main Power System including the Isolated Phase Bus and Main Power Transformer, and the Offsite Power System including the 230 kV switchyard and the CP&L transmission network (grid). The remainder of the Main Power System is addressed in BOP LR Section 2.4 "Turbine Generator Evaluations."

2.17.1.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the Onsite Electrical Distribution Systems provide AC or DC power to plant electrical loads at various voltage levels commensurate with the load requirements. The onsite AC power distribution system receives power under normal operating conditions through the Unit Auxiliary Transformers. Under start-up and shutdown conditions, power is supplied through Start-up Transformers. Four non-safety-related 6.9 kV switchgear buses (1A, 1B, ID and IE) provide the path of power from these transformers to the onsite power distribution system. Switchgear buses 1D and 1E provide the path of power to two independent safety-related switchgear buses 1A-SA and 1B-SB, which provide power to the redundant safety-related electrical loads. Should the preferred (offsite) power to these safety-related buses be unavailable, onsite power is supplied directly to the safety-related power distribution system from two emergency diesel generators (DG 1A-SA and DG 1B-SB). The Emergency Diesel Generator System is discussed in BOP Licensing Report Section 2.17.2.

The onsite DC Power System is designed to provide a source of reliable continuous power for the plant protection system, controls, instrumentation, and other DC loads for start-up, operation, and shutdown modes of plant operation.

The Onsite Electrical Distribution Systems are comprised of the following systems:

- 6.9 kV AC Distribution System
- 480V AC Distribution System
- 208/120 V AC Distribution System
- Non-Class 1E Uninterruptible AC System
- Class 1E Uninterruptible AC System
- 250 V DC Distribution System
- Class 1E 125 V DC Distribution System
- Non-Class 1E 125 V DC Distribution System

Safety-related electrical equipment is designated "Class 1E" and the safety-related electrical distribution system equipment, including the raceway system, is designed to meet seismic Category I requirements.

The Isolated Phase Bus is designed to deliver power from the generator terminals to the three single-phase main step-up transformers. The Main Power Transformer (MPT) bank consists of the three single-phase transformers and is connected wye on the high voltage side and delta on the low voltage side. The high voltage windings of the main transformer bank are connected to the 230 kV switchyard by overhead lines and circuit breakers. The low voltage windings are directly connected to the generator via the Isolated Phase Bus. These systems only function during normal operation.

The supply for preferred (offsite) power is the 230 kV system. Seven 230 kV transmission lines connect the switchyard to the transmission network. Each of the switchyard two 230 kV buses is a source of preferred power for the Unit. The 230 kV switchyard provides power through the Start-up transformers for the Unit's auxiliary systems for start-up, emergency or controlled shutdown, or when the power through the Unit's Auxiliary transformers is not available.

The 230 kV switchyard utilizes breaker-and-a-half and double-breaker schemes. The Unit is connected in a double-breaker scheme. The switchyard is provided with two independent 125V DC systems to furnish the control power for the circuit breakers. These systems are independent of the plant DC systems.

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change the design functions of the Onsite Electrical Distribution Systems. The existing Onsite Electrical Distribution Systems and Main Power System remain capable of satisfying regulatory commitments in accordance with the existing FSAR.

<u>Configuration Changes</u>:

There are no configuration changes associated with the Onsite Electrical Distribution Systems and Main Power System under the SGR/Uprate conditions.

<u>Revised Process Conditions</u>:

The evaluation of the Onsite Electrical Distribution System resulted in a net auxiliary load increase. This is due to the Non-Class 1E 6.9kV AC load increase as described in Table 2.17.1.1-1. The evaluation of the Turbine/Generator, Isolated Phase Bus, and MPTs used a net auxiliary load increase for conservatism. The load level changes to these components are tabulated in Table 2.17.1.1-2.

Table 2.17.1.1-1 - Non-Class 1E 6.9KV AC Loads Changes

					Approximate Pump HP Load Levels @ 100%		**Net System Load Change for all pumps**
	Mechanical Systems	**Qty**	**Rated HP Nameplate**	**Pump HP used in Analysis**	**Pre SGR/Uprate**	**Post SGR/Uprate**	**From Current to Uprate Operations**
1.	SG Fdwtr Pumps 1A&1B	2	9000	8770	6591	8287	+3392 HP
2.	Cond. Booster Pumps 1A&1B	2	3000	2780	2173	2905	+1464 HP
3.	Cond. Pumps 1A & 1B	2	2000	1818	1802	1830	+56 HP
4.	Reactor Coolant Pumps 1A, 1B, &1C	3	7000	7060	7060	7002	-174 HP
5.	Heater Drn. Pumps 1A & 1B	2	1500	1300	1175	1138	-74 HP
	Margin between analysis and uprate approximately 1190 HP						**+4664 HP**

Table 2.17.1.1-2 – Turbine Generator to Main Transformer Power Flow

	Pre-SGR/Uprate			**Post-SGR/Uprate**		
Units	MWe	Mvar	MVA	Mwe	Mvar	MVA
T/G Output Rating	950.1	433.3	1,044.2	998.0	309.9	1,045.0
Unit Auxiliary Load	78.6	51.6	94.0	77.6	51.0	92.9
Net to LV-side of MPTs	871.5	381.7	951.4	920.4	258.9	956.1
Estimated MPT Losses	3.0	122.7	122.8	3.0	122.7	122.8
Net out to HV-side of MPTs	868.5	259.0	906.3	917.4	136.2	927.4
MPT Rating			1,008.0			1,008.0

The impact of the load changes under the SGR/Uprate conditions, on the Onsite Electrical Distribution Systems and Main Power System, is addressed in Section 2.17.1.2.

2.17.1.2 Description of Analyses and Evaluations

The Onsite Electrical Distribution Systems and Main Power System were evaluated to ensure that, following the SGR/Uprate; the systems remain capable of performing required functions in accordance with the existing licensing bases specified in the FSAR.

The process information, tabulated in Tables 2.17.1.1-1, and 2.17.1.1-2, describes the load changes to the Onsite Electrical Distribution Systems and the MPTs. This information has been used to determine that the MPTs, Unit Auxiliary Transformers, Startup Transformers, Isolated Phase Bus (ampacity and cooling,) and large motors (RCP and condensate pump) will operate satisfactorily at the SGR/Uprate conditions, and to verify that all station auxiliary loads will continue to perform their intended safety-related and non safety-related functions at the SGR/Uprate conditions. Additionally, the impact of the SGR/Uprate upon the transmission system connected to the plant switchyard was evaluated for stability, thermal capacity, and voltage adequacy.

The evaluation revealed changes to the safety-related 6.9kV AC Distribution System due to SGR/Uprate, which are bounded by the existing design. As tabulated in Table 2.17.1.1-1, the SGR/Uprate changes will increase the load on the non-safety related 6.9 kV AC distribution system by ~4664 HP at 100% rated plant output.

For the other AC electrical distribution systems (i.e., the 480V AC, 208/120V AC, the Class 1E Uninterruptible AC, and the non-Class 1E Uninterruptible AC) and the DC distribution systems (i.e., 250V DC, Class 1E, 125V DC, and the non-Class 1E 125V DC), the evaluation identified no load changes due to SGR/Uprate.

The results of the analysis indicate that while the horsepower loads increase for the SGR/Uprate, they remain bounded and do not adversely impact the station onsite power system. Bounding steady-state voltages and motor starting voltages remain within acceptable limits. In addition, the rated nameplate horsepower for these pumps is greater than the horsepower load level at 100% power, as shown by Table 2.17.1.1-1. All other pump load levels are bounded by present conditions following SGR/Uprate.

The net load changes to the Onsite Electrical Distribution Systems, due to SGR/Uprate, show a total net load increase of ~4664 horsepower at 100% rated plant output. The system buses, breakers, and transformers (Startup and Unit Auxiliary) are bounded by the pre-SGR/Uprate analyzed conditions. Motor starting voltages, short circuit currents and bus and transformer relay settings are not adversely affected. The existing margins are not adversely impacted for the various supporting systems of the Onsite Electrical Distribution Systems.

The power flow from the Turbine/Generator to the MPTS is tabulated in table 2.17.1.1-2. SGR/Uprate conditions require an increase to the MPTs power level operation from the pre-SGR/Uprate level of 906.3 MVA to 927.4 MVA. The MPTs have a combined rating of 1008 MVA; therefore there is sufficient margin in the MPTs to handle the SGR/Uprate electrical power requirements.

There is no change to the output MVA level of the main generator therefore the Isolated Phase Bus is adequately rated for SGR/Uprate conditions.

Loadflow cases using scenarios from the FSAR were run with the existing as well as the SGR/Uprated plant output conditions. Case results show negligible difference in transmission system voltage levels by limiting the net MVAR range to 55-155 MVAR for the SGR/Uprated plant outputs. The line flows with the plant at SGR/Uprated power level, for both normal and contingency cases, are well within the line ratings. The results also show only a negligible difference between the present and post-SGR/Uprate range of switchyard and generator bus voltage levels with the net MVAR range of 55-155 MVAR after the SGR/Uprate.

Study cases show that plant switchyard voltage levels for the LOCA case and immediately after an unplanned Unit trip (non-LOCA case) under SGR/Uprate conditions will not be lower than 0.06% those at the present plant output levels. The switchyard voltages for an unplanned Unit trip are about 0.05% higher than voltages for a LOCA condition. Maintaining adequate voltage for a LOCA condition will ensure adequate switchyard voltage following a non-LOCA Unit trip.

Stability cases were run to check several enveloping conditions and scenarios as in the FSAR. Study cases show that the plant exhibits wide stability margin under SGR/Uprate conditions during a stuck-breaker type fault considering the 2 cycle, independent-pole-operated (IPO) 230 kV switchyard circuit breakers.

The rate of grid frequency decay for the SGR/Uprated Unit was checked by assuming that all the generation at Roxboro Plant is tripped while all of the ties in CP&L's Eastern System are also opened. The results indicate that the system oscillations are well damped and the frequency decay rate at the plant switchyard is well within the acceptable limit.

The results of the evaluation of the Onsite Electrical Distribution Systems and Main Power System and their components to satisfy individual design criteria and licensing bases as stated in the FSAR, are presented in Section 2.17.1.4. Acceptance criteria relevant to the Onsite Electrical Distribution Systems and Main Power System are identified in Section 2.17.1.3.

2.17.1.3 Acceptance Criteria

The design, operation, and functional capabilities for the SGR/Uprating program shall be bounded by the original system design conditions for the Onsite Electrical Distribution Systems and Main Power System.

The original system design parameters shall bound the revised system operating parameters for the SGR/Uprate program.

2.17.1.4 Results

The SGR/Uprate does not adversely affect the Onsite Electrical Distribution Systems and Main Power System. The existing Onsite Electrical Distribution Systems and Main Power System remain capable of satisfying regulatory commitments in accordance with the existing FSAR.

While there are net increases in auxiliary equipment system loads due to SGR/Uprate, the margins for the various supporting systems of the Onsite Electrical Distribution Systems are not adversely impacted. Post-SGR/Uprate Class 1E loads are bounded by the existing design. System reliability and capacity are not adversely affected.

Based on the Onsite Electrical Distribution Systems and Main Power System evaluation the SGR/Uprate does not adversely affect the accident analyses presented in the FSAR.

Pump motors are adequately sized for the SGR/Uprate conditions. Any changes to pump brake horsepower requirements are described on a system-by-system basis. There is no adverse impact on the individual motor or motor circuits.

Class 1E circuits will continue to be electrically isolated and physically separated from non-Class 1E circuits and devices. The SGR/Uprate does not modify or adversely affect existing power and control circuits that serve the Onsite Electrical Distribution Systems and Ma in Power System. Consequently, there are no electrical separation issues associated with the modifications to the Onsite Electrical Distribution Systems and Main Power System under the SGR/Uprate.

The Offsite Power System remains unchanged in characteristics and remains capable to support the plant Onsite Electrical Distribution System operations during normal and accident conditions after the SGR/Uprate implementation. It was also confirmed that the Offsite Power System voltages will not be adversely affected and will continue to support the Onsite Electrical Distribution System during normal and accident conditions after SGR/Uprate.

The evaluation indicates that there is no adverse impact to the existing Onsite Electrical Distribution Systems and Main Power System due to the effects of SGR/Uprate. Under SGR/Uprate conditions, the Onsite power systems will continue to provide independent, redundant and testable power supplies, each with its own distribution system capable of performing required safety functions.

2.17.1.5 Conclusions

The Onsite Electrical Distribution Systems and Main Power System are adequate for SGR/Uprate conditions, since their design capacities bound the SGR/Uprate conditions. The SGR/Uprate does not adversely affect system reliability and capacity. Therefore, the Onsite Electrical Distribution Systems and Main Power System will continue to meet required acceptance criteria and satisfy required functions, in accordance with the FSAR under SGR/Uprate conditions.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.17.1.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   - 3/4.8.1 Electrical Systems, "A.C. Sources"
   - 3/4.8.2 Electrical Systems, "D.C. Sources"
   - 3/4.8.3 Onsite Power Distribution
   - 3/4.8.4 Electrical Equipment Protective Devices

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

2.17.2 Emergency Diesel Generators

The Harris Nuclear Plant (HNP) Emergency Diesel Generators (EDGs) have been evaluated to determine their performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.17.2.1 Introduction and Background

The function of the EDGs is to provide a reliable source of alternate power to the emergency 6.9 kV buses for use in the event that normal sources of offsite power are not available. Each EDG is capable of supplying all power needed for the safe shutdown of the plant under design transient and accident situations. The EDGs automatically start either upon receipt of an Engineered Safety Feature Actuation Signal (ESFAS) or detection of a low bus voltage, as indicated by the bus undervoltage relays. An EDG automatically connects to its emergency bus when the generator output breaker closes, upon either low or loss of bus voltage.

The EDGs have been designed to be capable of continuous unattended operation at rated voltage and frequency in a range from 40 to 100 percent full load under emergency conditions for a minimum of 7 days. Each engine-generator is capable of being started and tested weekly under partial or full load to demonstrate its availability and functionality. The engine is also capable of running at idle speed for extended periods.

Each EDG is supported by ancillary systems (i.e., fuel oil, cooling water, lube oil, and starting air system). These ancillary systems are required for EDG operation. The EDG is supplied with fuel oil by the diesel generator fuel oil storage and transfer system. This subsystem maintains at least the minimum amount of fuel required for continuous EDG operation, at rated load, for 7 days. The cooling water system is designed to provide full load cooling to the diesel engines. The lube oil system provides lubrication to components of the diesel generators during all modes of operation. The starting air system provides compressed air to crank a cold diesel engine.

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change existing EDG design functions. The EDGs remain capable of satisfying regulatory commitments in accordance with the existing FSAR.

<u>Configuration Changes</u>:

There are no configuration changes associated with the EDGs under the SGR/Uprate conditions.

<u>Revised Process Conditions</u>:

The review of the EDGs did not identify any change to the ESF load list or in the EDG service environment due to the SGR/Uprate.

2.17.2.2 Description of Analyses and Evaluations

The EDGs were evaluated to ensure that, following the SGR/Uprate, they remain capable of performing required functions in accordance with the existing licensing bases specified in the FSAR.

To determine the capabilities of the EDGs to provide reliable and adequate power to all the required ESF loads for the worst case EDG loading condition, the existing design basis and capabilities were evaluated. Then the engineering evaluations of the NSSS, BOP and HVAC process systems and the accident analyses that were performed for the SGR/Uprate were reviewed. Next the worst case EDG loading conditions were established and the EDGs capability to satisfy these modified loading conditions was evaluated.

The EDG loads remain within the nameplate rating and are not affected by the SGR/Uprate. The maximum operating temperature and pressure for the EDG cooling water will also remain unchanged. The current supply pressure and maximum temperature of the EDG cooling water are 150 psig and 95°F, respectively, and are not affected by the implementation of the SGR/Uprate.

The fuel consumption of a diesel generator, at its rated load, is 445 gal/hr. Each fuel oil storage tank contains enough fuel to support continuous EDG operation, at its rated load, for seven days while also providing for testing in accordance with ANSI N195-1976. The SGR/Uprate will not affect the EDG fuel oil consumption rate or the required capacity of the fuel oil storage tanks.

The SGR/Uprate does not alter the basic EDG design functions. EDG load changes required to support the SGR/Uprate are bounded by the pre-SGR/Uprate conditions. Therefore, this can be accommodated by the existing design. The quantity of fuel required for continuous EDG operation, at rated load for seven days, remains unaffected by the SGR/Uprate. The design capabilities of the ancillary EDG subsystems are also unaffected by the SGR/Uprate.

The results of the evaluation of the EDGs and their individual components to satisfy applicable design and licensing bases in accordance with the FSAR are presented in Section 2.17.2.4. Acceptance criteria, relevant to the EDGs, are identified in Section 2.17.2.3.

2.17.2.3 Acceptance Criteria

The design, operation, and functional capabilities for the SGR/Uprating program shall be bounded by the original system design conditions for the Emergency Diesel Generator System.

The original system design parameters shall bound the revised system operating parameters for the SGR/Uprating program.

2.17.2.4 Results

No changes to EDG system configuration, ancillary systems, and service environment are needed to support the SGR/Uprate. The SGR/Uprate does not adversely affect EDG loads since changes are consistent with, or bounded by, the existing design. The SGR/Uprate does not impact the nameplate and brake horsepower of any ESF motor loads, the power requirements of ESF static loads and their loading sequence to the EDGs.

EDG ratings are sufficient to supply reliable power to all safety-related loads in their respective divisions as well as those non-safety related loads for which it is desirable to have manual loading capability on the EDGs. The SGR/Uprate evaluation ensures reliable and adequate power to all of the required ESF loads for the worst case EDG loading conditions. The EDG SGR/Uprate evaluation considered NSSS, BOP, and HVAC process systems including accident analyses. The evaluation also considered affects on the ancillary EDG systems and indicated that they are adequate for SGR/Uprate conditions, since their existing design capacities are conservative and bound the SGR/Uprate conditions.

The existing EDG rating is adequate. Diesel generator load profiles for the loss of offsite power (LOOP), loss of coolant accident (LOCA), and simultaneous LOOP and LOCA events have been reviewed to verify the continued capability of the EDGs to perform their intended safety function. No changes were identified that would adversely affect EDG loading, EDG environmental qualification, or service conditions. Also, the response time capability for diesel generator starting and loading is not impacted due to the SGR/Uprate.

Class 1E circuits will continue to be electrically isolated and physically separated from non-Class 1E circuits and devices. The SGR/Uprate does not modify or adversely affect existing power and control circuits that serve the EDGs, their ancillary systems, and the On-Site Emergency AC Power System. Consequently, there are no electrical separation issues associated with the modifications to the EDGs under the SGR/Uprate.

The SGR/Uprate will not affect the loading capacity of the EDG. The SGR/Uprate will not impact the nameplate and brake horsepower of all ESF motor loads, the power requirements of ESF static loads and their loading sequence to the EDGs.

2.17.2.5 Conclusions

The existing EDG design is adequate to support the SGR/Uprate. There are no required configuration changes to design of the system or its subsystems. The existing system components are adequate to meet the SGR/Uprate conditions. EDG load demands following the SGR/Uprate are consistent with, or bounded by the existing design. The SGR/Uprate does not adversely affect the ability of the EDGs, their ancillary systems, or the On-Site Emergency AC Power System to perform required functions in accordance with the FSAR. Following the SGR/Uprate, reserve capacity remains available for future loads, if necessary.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWT bound operation with the Delta 75 RSGs at the current NSSS thermal power level of 2787.4 MWt.

2.17.2.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.8.1 Electrical Systems, "A.C. Sources"

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

2.17.3 Station Blackout

The Harris Nuclear Plant (HNP) Station Blackout (SBO) provisions have been evaluated with respect to their adequacy in support of plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.17.3.1 Introduction and Background

SBO is defined in 10CFR50.2 as the complete loss of alternating current (AC) electric power to the essential and non-essential switchgear buses in a nuclear power plant. This entails a loss of normal off-site power (i.e., loss of the non-Class 1E, off-site, electric power supply concurrent with turbine trip in conjunction with the postulated unavailability of the Class 1E, on-site, emergency AC power system).

FSAR Section 8.3.1.2.21 indicates that the minimum acceptable SBO duration capability is based on the following factors:

- Redundancy of the on-site emergency AC power supply,
- Reliability of each of the on-site emergency power sources,
- Expected frequency of loss of off-site power, and
- Probable time needed to restore off-site power.

The FSAR also states procedures are to be utilized by the operators in coping with a SBO event. Procedures exist to:

- Recognize the SBO,
- Cope with the SBO,
- Restore off-site power,
- Restore power to the station's emergency buses, and
- Allow station recovery from the effects of the SBO.

The FSAR states that the EDG target reliability is 0.95. Emergency Diesel Generator (EDG) design, operational testing, surveillance, and periodic maintenance are relied on to ensure an EDG target reliability of 0.95.

Finally, FSAR Section 8.3.1.2.21 identifies the following system characteristics as being relevant to SBO coping capability:

- Condensate inventory for decay heat removal,
- Class 1E battery capacity,
- Compressed air,
- Effects of loss of ventilation,
- Containment isolation, and
- Reactor Coolant inventory.

Configuration Changes:

The evaluation of the plant's ability to cope with a postulated SBO, following the SGR/Uprate, requires no configuration changes to the components and functions credited in the SBO coping assessments.

Revised Process Conditions:

The SGR/Uprate involves increases in the mass and energy stored in the Reactor Coolant System (RCS) and the RSGs.

2.17.3.2 Description of Analyses and Evaluations

The HNP was evaluated to ensure that, following the SGR/Uprate, systems remain capable of performing required functions, in the event of SBO, in accordance with applicable licensing bases specified in the FSAR.

The total stored energy in the RCS and Steam Generators, under SGR/Uprate conditions, was used to determine the required Condensate Storage Tank (CST) inventory for SBO.

The results of the SBO evaluation to satisfy applicable design and licensing bases in accordance with the FSAR are presented in Section 2.17.3.4. Acceptance criteria, relevant to the SBO, are identified in Section 2.17.3.3.

2.17.3.3 Acceptance Criteria

The applicable acceptance criteria are 10CFR50.63 "Loss of All Alternating Current Power". Final Safety Analysis Report (FSAR) Section 8.3.1.2.21 describes compliance with NRC guidance contained in Regulatory Guide 1.155, "Station Blackout".

CP&L also follows the guidance of NUMARC 87-00 as accepted by RG 1.155.

2.17.3.4 Results

The total stored energy in the RCS and SGs increased due to SGR/Uprating.

The total condensate required to cooldown the RCS during the 4 coping period hours after SBO has increased to approximately 112,200 gallons. The usable volume of the CST is 238,000 gallons. Therefore, there is more than sufficient condensate water available from the CST to accommodate the decay heat removal during a SBO event for the 4 hour coping period.

Specific Systems are evaluated as follows:

1. Class 1E & Non-Class 1E Battery systems - There are no identified changes in configuration and/or load changes to the Class 1E & Non-Class 1E Battery Systems due to SGR/PUR.

- Compressed Air (including other gases) Systems - There are no identified changes in configuration or requirements to Compressed Air (including other gases) Systems.

- Loss of HVAC Systems - There are no identified changes in equipment cooling requirements.

- Containment Isolation - There are no identified changes in configuration or requirements to Containment Isolation integrity.

The coping period was selected independent at the plants power output; the coping period for SGR/Uprate remains constant at 4 hours. SGR/Uprate does not impact the HNP EDG target reliability of 0.95.

2.17.3.5 Conclusions

Based on the evaluation presented above, the SGR/Uprate does not adversely affect the ability of the plant to mitigate a postulated SBO event in accordance with the existing FSAR licensing basis.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation of the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.17.3.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications

3. RG 1.155, "Station Blackout," (Rev. 0, 8/88)

4. 10CFR50.2, "Definitions"

5. 10CFR50.63, "Loss of All Alternating Current Power"

6. Nuclear Management and Resources Council (NUMARC) "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors" NUMARC-8700, November 1987

7. NUMARC 87-00, "Supplemental Questions/Answers and Major Assumptions," December 27, 1989

2.18 Miscellaneous Mechanical Reviews

MOVs Program Impacts Due to SGR/Uprate

MOV design differential pressures are developed from conservative, limiting conditions such as pump shutoff pressure, maximum/minimum tank levels, maximum sump levels, and peak containment pressure. As described in the BOP and NSSS LRs, the CCW System is the only system with a physical modification that impacts safety system performance. In the Component Cooling Water System (CCW), revised pump performance is required. Refer to BOP Section 2.6 (CCW) for additional information on these changes.

A modification to the Component Cooling Water pumps increases flow and consequently changes shut-off pressure. The modification process includes reviews for changes to valves in the MOV program. Changes, as necessary, are incorporated into calculations and MOV configuration.

The SGR/Uprate analyses have identified the following impacts on MOV evaluations:

a. Containment peak pressure increases (from 41.2 psig to 41.8 psig)
b. Containment sump level changes (from 228.3 feet to 228.5 feet)
c. Steam Generator pressures (during actuation of the AFW isolation to a faulted Steam Generator).

The modification phase of the project addresses the details of these impacts to MOV evaluations.

2.19 Miscellaneous I&C Reviews

The Harris Nuclear Plant (HNP) Miscellaneous Instrumentation and Controls (I&C) reviews were conducted for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power of 2912.4 MWt.

Calibration and sealing of instruments were evaluated relative to process measurement conditions and current setpoint requirements. Although setpoint changes/rescaling (including instrument replacements, as necessitated by new range requirements) will be required for SGR/Uprate implementation (e.g., NSSS LR Section 4.3, 6.7, and 4.1.6), the SGR/Uprate does not change I&C systems/equipment design functions. Radiation monitoring functions were evaluated relative to changes in SGR/Uprate radiation source terms; except for a recommended adjustment in the RAB exhaust monitor alarm setpoint, radiation monitor response and existing setpoints are adequate with respect to these source term changes. Therefore, the I&C systems/equipment will remain capable of satisfying regulatory commitments in accordance with the existing FSAR.

2.20 Radioactive Waste Management

2.20.1 Solid and Liquid Waste Processing Systems

The Harris Nuclear Plant Solid and Liquid Waste Processing Systems (S&LWPS) have been evaluated to determine their performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.20.1.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the Solid Waste Processing System (SWPS) is designed to collect, control, process, package, handle and temporarily store radioactive waste generated as a result of normal operation of the plant.

The Liquid Waste Processing System (LWPS) provides for the collection, storing, processing, and controlled release of radioactive and potentially radioactive liquids associated with the operation of the nuclear power plant. The discharge of treated wastes is controlled and monitored to ensure that any discharges are as low as reasonable achievable (ALARA) and that they are in conformance with the requirements specified in 10CFR20 and 10CFR50, Appendix I.

The LWPS consists of the following subsystems: 1) Equipment Drain Treatment System, 2) Floor Drain Treatment System, 3) Laundry and Hot Shower Treatment System, 4) Chemical Drains, 5) Filter Backwash System, and 6) Secondary Waste Treatment System.

Collectively these systems are referred to below as the Solid and Liquid Waste Processing Systems (S&LWPS).

<u>Configuration Changes</u>:

There are no configuration changes associated with the S&LWPS under the SGR/Uprate conditions.

<u>Revised Operating Parameters</u>:

There are no process parameter changes in the S&LWPS resulting from SGR/Uprate.

2.20.1.2 Description of Analyses and Evaluations

The S&LWPS were evaluated to ensure that, following the SGR/Uprate, the systems remain capable of performing required functions in accordance with the existing licensing bases specified in the FSAR.

Evaluations of the S&LWPS have been revised as appropriate to reflect SGR/Uprate NSSS operating parameter changes. The PWR-GALE computer code was used to provide revised input to the tables in FSAR, by calculating radionuclide concentrations and the releases of radioactive material in gaseous and liquid effluents. The LADTAP II code was used to calculate individual doses from expected liquid radioactive releases. These programs are accepted for use by the NRC.

Liquid releases to the environment result in doses to the population around the site. These doses are estimated using methods presented in Regulatory Guide (RG) 1.109, Rev. 1, and RG 1.113, Rev. 1.

The updated calculations for the LWPS result in revised radioactive liquid releases (refer to revised attached FSAR Table 11.2.3-5). The revisions stem from changes in operating parameters as well as changes resulting from the SGR/Uprate. The major changes include:

1. Increasing the normal letdown flowrate from 60 to 106 gpm ;

2. Changes in the steam/water masses for the RCS and Steam Generator, and

3. Use of the HNP historical data for radwaste stream volumes from 1998 and 1999, which are substantially reduced from the values currently used in the FSAR analysis.

Since the previous analysis was done for a core power of 2900 MWt, the changes in results are due to the changes summarized above.

The results of the evaluation of the S&LWPS and their individual components to satisfy applicable design and licensing bases in accordance with the FSAR are presented in Section 2.20.1.4. Acceptance criteria, relevant to the S&LWPS, are identified in Section 2.20.1.3.

2.20.1.3 Acceptance Criteria

The licensing bases for the S&LWPS are described in the FSAR. The acceptance criteria relevant to SGR/Uprate are:

1. Regulatory limits (10CFR Part 20 and 10CFR Part 50 Appendix I) continue to be met for releases of radioactive effluents.

The design, operation, and functional capabilities of the S&LWPS described in the FSAR, and as affected by the SGR/Uprate, were evaluated against the acceptance criteria. The results of these evaluations are described in Section 2.20.1.4.

2.20.1.4 Results

There are no configuration changes to the system design as a result of SGR/Uprate. The existing S&LWPS design is adequate for the SGR/Uprate.

Radioactive releases and doses resulting from plant operation have decreased compared to current FSAR analysis. The revised individual doses from liquid radioactive releases (i.e., FSAR Table 11.2.3-4 attached to this report) are within the limits set forth by 10CFR50, Appendix I. The revised Concentrations of Radionuclides in Liquid Effluents (i.e., FSAR Tables 11.2.3-3 and 11.2.3-5 attached to this report) are within the limits set forth by 10CFR20.

2.20.1.5 Conclusions

The SGR/Uprate does not change S&LWPS design functions. There are no configuration changes associated with the S&LWPS under SGR/Uprate conditions. Changes to NSSS of the process conditions are minor and within the capabilities of the existing design of the S&LWPS. Although there are changes to the values in the FSAR (Sections 11.1, 11.2, and 11.4) Tables, the S&LWPS remain capable of satisfying regulatory requirements identified in Section 2.20.1.4.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.20.1.6 References

1. HNP Final Safety Analysis Report

2. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

3. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

4. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

5. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

6. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

7. NUREG-0017, Revision 1, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," dated April 1985

8. RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," (Rev. 1, 10/77)

9. RG 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," (Rev. 1, 4/77)

10. 10CFR50, Appendix I "Numerical Guides For Design Objectives And Limiting Conditions For Operation To Meet The Criterion "As Low As Reasonably Achievable" For Radioactive Material In Light-Water-Cooled Nuclear Power Reactor Effluents."

Table 2.20.1-1
Revision to FSAR Table 11.2.3-3
Normal Operational Concentrations of Radionuclides In Liquid effluents.

NUCLIDE	COOLING TOWER DISCHARGE		AVERAGE RESERVOIR	
	[uCi/ml]	[Conc./ EC limit] (1)	[uCi/ml]	[Conc./ EC limit] (1)
H 3	6.38E-05	6.38E-02	2.18E-05	2.18E-02
Na 24	2.19E-11	4.38E-07	9.64E-15	1.93E-10
Cr 51	3.06E-12	6.12E-09	5.70E-14	1.14E-10
Mn 54	1.81E-12	6.04E-08	2.49E-13	8.28E-09
Fe 55	1.38E-12	1.38E-08	3.36E-13	3.36E-09
Co 58	4.97E-12	2.48E-07	2.21E-13	1.10E-08
W187	1.28E-12	4.25E-08	8.94E-16	2.98E-11
Np239	1.27E-12	6.34E-08	2.09E-15	1.05E-10
Y 93	1.72E-12	8.58E-08	5.14E-16	2.57E-11
Mo 99	3.82E-12	1.91E-07	7.47E-15	3.74E-10
Tc 99m	3.31E-12	3.31E-09	5.83E-16	5.83E-13
Ru103	7.67E-12	2.56E-07	1.99E-13	6.65E-09
Ru106	1.02E-10	3.42E-05	1.58E-11	5.28E-06
Ag110m	1.47E-12	2.44E-07	1.78E-13	2.97E-08
Te131m	7.94E-13	9.92E-08	6.98E-16	8.72E-11
I131	9.30E-09	9.30E-03	5.20E-11	5.20E-05
Te132	1.03E-12	1.15E-07	2.35E-15	2.61E-10
I132	3.63E-11	3.63E-07	2.45E-15	2.45E-11
I133	9.09E-10	1.30E-04	5.60E-13	7.99E-08
I134	2.06E-12	5.16E-09	5.33E-17	1.33E-13
Cs134	2.80E-10	3.11E-04	6.21E-11	6.90E-05
I135	1.72E-10	5.75E-06	3.39E-14	1.13E-09
Cs136	1.62E-11	2.70E-06	1.45E-13	2.42E-08
Cs137	3.75E-10	3.75E-04	1.37E-10	1.37E-04
Ba140	1.08E-11	1.35E-06	9.51E-14	1.19E-08
La140	1.85E-11	2.06E-06	2.17E-14	2.41E-09
Ce143	1.47E-12	7.36E-08	1.43E-15	7.14E-11
Ce144	4.43E-12	1.48E-06	5.83E-13	1.94E-07
TOTAL Conc. / EC limit		7.39E-02		2.21E-02

(1) Note: Effluent Concentrations (EC) are based on the 1993 revision of 10CFR20

Table 2.20.1-2

Revision to FSAR Table 11.2.3-4

INDIVIDUAL DOSES FROM LIQUID RADIOACTIVE RELEASES

"AS LOW AS REASONABLY ACHIEVABLE" LOCATION

LIMITING CONDITIONS

ADULT DOSES

	DOSE (MREM PER YEAR INTAKE)
PATHWAY	**TOTAL BODY**
FISH	**7.66E-01**
DRINKING	**5.36E-02**
SHORELINE	**8.50E-04**
SWIMMING	**4.27E-05**
BOATING	**4.89E-05**
TOTAL	**8.20E-01**

TEENAGER DOSES

	DOSE (MREM PER YEAR INTAKE)
PATHWAY	**LIVER**
FISH	**1.07E+00**
DRINKING	**3.81E-02**
SHORELINE	**4.75E-03**
SWIMMING	**6.40E-05**
BOATING	**7.12E-05**
TOTAL	**1.12E+00**

Note: The maximum total body dose is to adults, and the maximum organ dose is to teenagers

Table 2.20.1-3
Revision to FSAR Table 11.2.3-5
Design Basis Concentrations of Radionuclides In Liquid Effluents.

NUCLIDE	EFFLUENT RELEASE	COOLING TOWER DISCHARGE		AVERAGE RESERVOIR DISCHARGE	
	[Ci/y]	[uCi/ml]	[Conc./EC limit]	[uCi/ml]	[Conc./EC limit]
H 3	3.29E+03	1.03E-04	1.03E-01	7.64E-05	7.64E-02
Na 24	3.23E-04	1.01E-11	2.03E-07	9.64E-15	1.93E-10
Cr 51	1.33E-04	4.19E-12	8.38E-09	1.69E-13	3.37E-10
Mn 54	1.12E-05	3.52E-13	1.17E-08	1.04E-13	3.48E-09
Fe 55	6.53E-05	2.05E-12	2.05E-08	1.08E-12	1.08E-08
Co 58	3.99E-04	1.25E-11	6.27E-07	1.20E-12	6.02E-08
W187	1.88E-05	5.91E-13	1.97E-08	8.94E-16	2.98E-11
Np239	1.87E-05	5.87E-13	2.94E-08	2.09E-15	1.05E-10
Y 93	2.26E-06	7.10E-14	3.55E-09	4.59E-17	2.30E-12
Mo 99	6.50E-03	2.04E-10	1.02E-05	8.63E-13	4.31E-08
Tc 99m	5.28E-03	1.66E-10	1.66E-07	6.31E-14	6.31E-11
Ru103	8.18E-06	2.57E-13	8.56E-09	1.44E-14	4.81E-10
Ru106	3.44E-06	1.08E-13	3.60E-08	3.60E-14	1.20E-08
Ag110m	2.87E-05	9.02E-13	1.50E-07	2.37E-13	3.95E-08
Te131m	1.85E-04	5.80E-12	7.25E-07	1.10E-14	1.38E-09
I131	8.49E+00	2.67E-07	2.67E-01	3.22E-09	3.22E-03
Te132	2.97E-03	9.33E-11	1.04E-05	4.59E-13	5.10E-08
I132	6.41E-03	2.01E-10	2.01E-06	2.94E-14	2.94E-10
I133	9.75E-01	3.06E-08	4.37E-03	4.07E-11	5.82E-06
I134	4.76E-05	1.50E-12	3.74E-09	8.35E-17	2.09E-13
Cs134	1.39E+00	4.36E-08	4.84E-02	2.09E-08	2.33E-02
I135	2.13E-02	6.70E-10	2.23E-05	2.85E-13	9.48E-09
Cs136	1.33E+00	4.18E-08	6.96E-03	8.07E-10	1.35E-04
Cs137	1.46E+00	4.60E-08	4.60E-02	3.63E-08	3.63E-02
Ba140	4.83E-05	1.52E-12	1.90E-07	2.89E-14	3.61E-09
La140	1.47E-05	4.63E-13	5.14E-08	1.17E-15	1.30E-10
Ce143	4.17E-06	1.31E-13	6.54E-09	2.74E-16	1.37E-11
Ce144	7.57E-06	2.38E-13	7.93E-08	6.76E-14	2.25E-08
TOTAL Conc. / EC limit			**4.76E-01**		**1.39E-01**

Note: Effluent Concentrations (EC) are based on the 1993 revision of 10CFR20

2.20.2 Gaseous Waste Processing System

The Harris Nuclear Plant (HNP) Gaseous Waste Processing System (GWPS) has been evaluated to determine its performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.20.2.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the GWPS is designed to collect, process and store gaseous wastes generated due to plant operations including anticipated operational occurrences. The system is designed to ensure that the release of gaseous effluents from the plant and expected offsite doses are as low as reasonably achievable.

The numerical values for plant releases and off site doses include consideration of the performance of the RAB HVAC system, the Containment Normal Purge System, and the Fuel Handling Building HVAC System.

Analysis of accident doses resulting from failure of a waste gas decay tank is addressed in BOP LR Section 2.22 "Personnel Radiation Dose Analysis."

All equipment in the GWPS is located in the Waste Processing Building (WPB) and controlled from the WPB Control Room. The GWPS operates in a closed loop using a waste gas compressor, catalytic recombiner and waste gas decay tank to collect fission product gases.

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change GWPS design functions. The existing GWPS remains capable of satisfying regulatory commitments in accordance with the existing FSAR.

<u>Configuration Changes</u>:

There are no configuration changes associated with the GWPS under the SGR/Uprate conditions.

<u>Revised Operating Parameters</u>:

There are no changes to the operating parameters of the GWPS.

2.20.2.2 Description of Analyses and Evaluations

The GWPS was evaluated to ensure that, following the SGR/Uprate, the system remains capable of performing required functions in accordance with existing licensing bases specified in the FSAR.

The PWR-GALE computer code was used to calculate the releases of radioactive material in gaseous and liquid effluents and the GASPAR computer code was utilized to determine the impacts due to the release of radioactive material to the atmosphere during normal operation for the SGR/Uprate conditions. These programs are industry standards and accepted for use by the NRC.

The changes to the gaseous releases result from changes from the past operating parameters, as well as changes resulting from the SGR/Uprate. The major changes include:

1. Increasing the normal letdown flowrate from 60 to 106 gpm (which reduced steady state primary coolant activities),

2. Changes in the steam/water masses for the RCS (increase) and Steam Generator (decrease),

3. Correction of the charcoal filter efficiencies for normal ventilation exhaust units (95%⇒ 90%) and correction to delete charcoal and HEPA filters on FHB normal exhaust.

4. The Condenser Vacuum Pump exhaust - as a conservatism it was assumed that it was not filtered.

5. Use of radwaste stream volumes from 1998 and 1999, which are substantially reduced from the earlier values.

The previous analysis was done for a core power of 2900 MWt. The above changes to operating parameter assumptions are all changes in interfacing systems, not changes in GWPS parameters which remain unchanged.

The GWPS evaluation demonstrates that the existing design meets all system functional requirements at SGR/Uprate conditions.

The results of the evaluation of the GWPS and its individual components to satisfy applicable design and licensing bases in accordance with the FSAR are presented in Section 2.21.2.4. Acceptance criteria, relevant to the GWPS, are identified in Section 2.20.2.3.

2.20.2.3 Acceptance Criteria

The licensing bases for the GWPS are described in the FSAR. The GWPS is designed to collect, process and store gaseous wastes ensuring that the release of the gaseous effluents from the plant and expected offsite doses are as low as reasonable achievable, and that they are in conformance with the requirements specified in 10CFR20 and 10CFR50.

In accordance with the guidance of Section 5.6 of NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," the curie content of each of the Gas Storage Tanks is limited to the quantity of radioactivity which would result in an offsite dose of 500 mrem to the whole body, following an uncontrolled release. Technical Specification 6.8.4.j requires a "Gas Storage Tank Radioactivity Monitoring Program," to ensure that this limit is not exceeded.

10CFR20 requires that the sum of all the ratios of radionuclide concentrations to effluent concentration limits in the gaseous effluents must be less than 1.0 for the design basis and normal operation. Only the normal operating concentrations were recalculated. This information can be found in the attached revised FSAR Table 11.3.3-3. 10CFR50, Appendix I requires that the annual individual doses from gaseous radioactive releases must be less than 15 mrem to any organ and 5 mrem to the total body. This information can be found in the attached revised FSAR Table 11.3.3-4.

The design, operation, and functional capabilities of the GWPS described in the FSAR, and as affected by the SGR/Uprate, were evaluated against the acceptance criteria. The results of these evaluations are described in Section 2.20.2.4.

2.20.2.4 Results

The existing GWPS design is adequate for the SGR/Uprate. Maximum doses associated with calculated gaseous radiological releases will decrease compared to current FSAR analysis. There are no configuration changes to the systems design as a result of SGR/Uprate. The revised individual doses from gaseous radioactive releases (i.e., FSAR Table 11.3.3-4 attached to this report) are within the limits set forth by 10CFR50 Appendix I. The revised Concentrations of Radionuclides in Liquid Effluents (i.e., FSAR Table 11.3.3-3 attached to this report) are within the limits set forth by 10CFR20. The existing system components are adequate to meet the SGR/Uprate. GWPS radiological protection functions and capabilities are not adversely affected and remain in accordance with the existing FSAR.

The maximum quantity of radioactivity contained in each gas storage tank, based on a 1% fuel clad failure, is such that in the event of an uncontrolled release of the tanks contents, the offsite dose would be less than 500 mrem. Refer to BOP LR Section 2.22 for the results of the accidental release of a Waste Gas Decay Tank (WGDT).

The evaluation indicates that the SGR/Uprate does not adversely affect the ability of the GWPS to continue to satisfy required functions, including control of radioactive effluent release.

2.20.2.5 Conclusions

The SGR/Uprate does not change GWPS design functions. There are no configuration changes associated with the GWPS under SGR/Uprate conditions and changes to GWPS process conditions are minor and within the capabilities of the existing design. Although there are changes to the values in the FSAR Tables, the GWPS remain capable of satisfying regulatory requirements in accordance with the existing FSAR.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.20.2.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.11.2 Gaseous Effluents
   6.8.4.j Gas Storage Tank Radioactivity Monitoring Program

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1" dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"

9. NUREG-0017, Revision 1, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," dated April 1985

10. RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," (Rev. 1)

11. NUREG-0133, Section 5.6, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants"

12. Branch Technical Position ESTB 11-5, "Postulated Radioactive Releases due to a Waste Gas System Leak or Failure"

13. 10CFR20, "Standards for Protection Against Radiation"

14. 10CFR50.34, "Contents of Applications; Technical Information"

15. 10CFR50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents"

Table 2.20.2-1

Revised FSAR TABLE 11.3.3-3

ANNUAL AVERAGE CONCENTRATION OF RADIOACTIVITY AT THE SITE BOUNDARY- ONE UNIT- NORMAL OPERATION

		(A)	(B)	(C)
	Release gale-gas[1] Ci/yr	Concentration UCi/ml	10CFR20[2] EC limit (uCi/ml)	Ratio Conc./EC limit
KR85M	6.10E+01	1.17992E-11	1.00E-07	1.18E-04
KR85	5.90E+02	1.14124E-10	7.00E-07	1.63E-04
KR87	2.50E+01	4.83574E-12	2.00E-08	2.42E-04
KR88	8.10E+01	1.56678E-11	9.00E-09	1.74E-03
XE131M	1.40E+02	2.70802E-11	2.00E-06	1.35E-05
XE133M	3.10E+01	5.99632E-12	6.00E-07	9.99E-06
XE133	7.50E+02	1.45072E-10	5.00E-07	2.90E-04
XE135M	8.00E+00	1.54744E-12	4.00E-08	3.87E-05
XE135	4.40E+02	8.51091E-11	7.00E-08	1.22E-03
XE138	7.00E+00	1.35401E-12	2.00E-08	6.77E-05
CR-51	2.70E-04	5.2226E-17	3.00E-08	1.74E-09
MN-54	3.50E-04	6.77004E-17	1.00E-09	6.77E-08
CO-57	8.10E-06	1.56678E-18	9.00E-10	1.74E-09
CO-58	2.10E-02	4.06202E-15	1.00E-09	4.06E-06
CO-60	8.20E-03	1.58612E-15	5.00E-11	3.17E-05
FE-59	2.80E-05	5.41603E-18	7.00E-10	7.74E-09
SR-89	2.20E-03	4.25545E-16	2.00E-10	2.13E-06
SR-90	8.60E-04	1.6635E-16	6.00E-12	2.77E-05
ZR-95	1.40E-05	2.70802E-18	5.00E-10	5.42E-09
NB-95	2.40E-03	4.64231E-16	2.00E-09	2.32E-07
RU-103	5.40E-05	1.04452E-17	9.00E-10	1.16E-08
RU-106	6.90E-05	1.33467E-17	2.00E-11	6.67E-07
SB-125	5.70E-05	1.10255E-17	7.00E-10	1.58E-08
CS-134	1.70E-03	3.28831E-16	2.00E-10	1.64E-06
CS-136	3.30E-05	6.38318E-18	9.00E-10	7.09E-09
CS-137	2.80E-03	5.41603E-16	2.00E-10	2.71E-06
BA-140	4.20E-06	8.12405E-19	2.00E-09	4.06E-10
CE-141	1.40E-05	2.70802E-18	8.00E-10	3.39E-09
I-131	1.30E-02	2.51459E-15	2.00E-10	1.26E-05
I-133	4.80E-02	9.28463E-15	1.00E-09	9.28E-06
H-3	2.10E+02	4.06202E-11	1.00E-07	4.06E-04
C-14	7.30E+00	1.41204E-12	3.00E-09	4.71E-04
AR-41	3.40E+01	6.57661E-12	1.00E-08	6.58E-04

Notes:

(1) GALE output, attachment C, Gaseous Releases with Charcoal Efficiency of 90%

(2) Effluent Concentrations [EC] limits are based on the 1993 revision to 10CFR20.

Table 2.20.2-2
Revised FSAR TABLE 11.3.3-4

POTENTIAL DOSES FROM GASEOUS RADIOACTIVE RELEASES- ONE UNIT- NORMAL OPERATION

Site Boundary Air Doses[2]	(mrad/year)
Gamma	3.49E-01
Beta	6.31E-01

	Maximum Individual Doses (mrem/year)			
Total Body	Adult (17+ yrs)	Teenagers (11-17 yrs)	Children (1-11 yrs)	Infant (0-1 yrs)
Immersion [3]	7.00E-02	7.00E-02	7.00E-02	7.00E-02
Inhalation [3]	1.90E-02	1.91E-02	1.69E-02	9.66E-03
Ground Deposition [3]	2.75E-02	2.75E-02	2.75E-02	2.75E-02
Vegetables [4]	1.52E-01	2.26E-01	4.90E-01	
Milk [5]	3.21E-02	5.44E-02	1.24E-01	2.49E-01
Meat [3]	3.97E-02	3.21E-02	5.82E-02	
TOTAL	3.40E-01	4.29E-01	7.87E-01	3.56E-01
SKIN				
Immersion [3]	2.15E-01	2.15E-01	2.15E-01	2.15E-01
Inhalation [3]	1.82E-02	1.86E-02	1.65E-02	9.47E-03
Ground Deposition [3]	3.23E-02	3.23E-02	3.23E-02	3.23E-02
Vegetables [4]	1.29E-01	1.98E-01	4.46E-01	
Milk [5]	3.09E-02	5.32E-02	1.23E-01	2.48E-01
Meat [3]	3.90E-02	3.18E-02	5.78E-02	
TOTAL	4.64E-01	5.49E-01	8.91E-01	5.05E-01
THYROID				
Immersion [3]	7.00E-02	7.00E-02	7.00E-02	7.00E-02
Inhalation [3]	3.41E-02	3.85E-02	4.02E-02	3.13E-02
Ground Deposition [3]	2.75E-02	2.75E-02	2.75E-02	2.75E-02
Vegetables [4]	1.50E-01	2.16E-01	4.73E-01	
Milk [5]	3.60E-02	6.13E-02	1.39E-01	2.89E-01
Meat [3]	4.11E-02	3.32E-02	6.00E-02	
TOTAL	3.59E-01	4.47E-01	8.10E-01	4.18E-01

(1) All dose calculated at the critical receptor location.
(2) Calculated at 1.36 miles in the S direction using meteorological dispersion from ODCM Revision 3.
(3) Calculated at 1.8 miles in the NNE direction using meteorological dispersion from ODCM Revision 3.
(4) Calculated at 1.7 miles in the NNE direction using meteorological dispersion from ODCM Revision 3.
(5) Calculated at 2.2 miles in the N direction using meteorological dispersion from ODCM Revision 3.

2.21 Process Sampling System

The Harris Nuclear Plant (HNP) Process Sampling Systems have been evaluated to determine their performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.21.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the Process Sampling System is comprised of the Primary Sampling System (PSS), the Secondary Sampling System (SSS), and the Post Accident Sampling System (PASS). The PSS provides liquid and gaseous samples during normal operation.

The PSS provides primary-side liquid and gaseous samples in the RAB for analysis to determine fission and corrosion product activity levels, boron concentration, lithium, pH and conductivity levels, radiation levels, crud concentration, dissolved gas concentration, chloride concentration, and gas composition in various tanks. Samples taken from the PSS pass through a run of piping and tubing which ensure a minimum decay time of 28 seconds and components that reduce the temperature and pressure.

The SSS provides secondary-side liquid and gaseous samples during normal operation. The SSS provides a means for continuous monitoring of liquid and steam purity in the condensate, heater drains, feedwater, steam cycle, steam generator blowdown, condensate storage tank, and main steam systems.

PASS provides a capability to obtain and quantitatively analyze reactor coolant gas and liquid samples during and following an accident to determine the presence and amount of core degradation.

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change Process Sampling System design or functions. The existing Process Sampling System remains capable of satisfying regulatory commitments under the SGR/Uprate conditions.

Configuration Changes:

There are no configuration changes associated with the Process Sampling System under the SGR/Uprate conditions.

Revised Process Conditions:

While the normal operating conditions for SGR/Uprate results in slightly higher temperatures and slightly higher pressures on the secondary side on the Steam Generators, the no-load conditions on the secondary side and operating conditions for the Reactor Coolant System sample points bound any of the projected increased sample conditions for SGR/Uprate.

2.21.2 Description of Analysis and Evaluations

The RCS Sampling System was evaluated to ensure that, following the SGR/Uprate, the system remains capable of performing required functions in accordance with existing design/operating limits.

The Process Sampling System was evaluated with regard to process changes resulting from the SGR/Uprate. These changes involve process line conditions and their effect on system piping and components from the point of sample withdrawal up to the analyzing station, sampling station, or local sampling point. Consideration was also given to the increased radiological doses in the post-accident service environments. Primary and secondary temperatures and pressures for SGR/Uprate conditions were compared to the original design and operating values, as described in DBD-101 (RCS) and DBD-101A (Secondary).

The Process Sampling System and its functions are not changed by SGR/Uprate. These changes in process operating conditions are within the system design limits; existing piping is adequate for the process condition changes. The actual samples will be cooled and the pressure reduced to 30 psig consistent with the current practice. Design and maximum operating temperatures and pressures for the PSS, SSS, and PASS were evaluated.

The SGR/Uprate will result in more severe conditions within the RCB and RAB following a design basis accident, during which PASS is required. Although the post-accident radiological conditions are more severe under SGR/Uprate, these post-accident conditions do not adversely affect the capability of PASS components from performing required functions in accordance with the FSAR. PASS shielding remains adequate for required post accident access.

2.21.3 Acceptance Criteria

The licensing bases for the Process Sampling System are described in FSAR Section 9.3.2. The PSS and SSS provide a means for continuous monitoring and control of the reactor coolant chemistry and the purity of the condensate and feedwater to the steam generator. The PASS is designed to sample reactor coolant during post-accident conditions.

The design, operation, and functional capabilities of the Process Sampling System described in the FSAR, and as affected by the SGR/Uprate, were evaluated against the design bases.

2.21.4 Results

The existing Process Sampling System remains capable of satisfying design requirements.

The SGR/Uprate does not require any configuration changes that involve the Process Sampling System.

The increased NSSS thermal output under the SGR/Uprate results in a minor increase in the maximum PSS sample temperature for RPV outlet temperature and a minor increase in the maximum SSS sample pressure for SG pressure. Existing piping is adequate for the process condition changes.

2.21.5 Conclusions

The SGR/Uprate does not adversely affect the Process Sampling System. There are no configuration changes associated with the SGR/Uprate. The minor increase in the maximum Primary Sampling System (PSS) sample temperature for RPV outlet temperature and a minor increase in the maximum Secondary Sampling System (SSS) sample pressure for SG pressure resulting from SGR/Uprate are within the system design limits. Although the post-accident radiological conditions are more severe under SGR/Uprate, these post-accident conditions do not adversely affect the capability of Process Sampling System components from performing required functions. The Process Sampling System remains capable of performing required functions, normal and post-accident under SGR/Uprate conditions, in accordance with the existing design bases.

The results obtained with the Delta 75 RSG at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSG at the current NSSS power level of 2787.4 MWt.

2.21.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.4.7 RCS - Chemistry
   3/4.4.8 RCS - Specific Activity
   3/4.7.1.4 Plant Systems – Specific Activity

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"

9. RG 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants To Assess the Plant and Environs Conditions During and Following an Accident," (Rev. 3, 5/83)

10. 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants"
    Criterion 1 "Quality Standards and Records"
    Criterion 2 "Design Bases for Protection Against Natural Phenomena"
    Criterion 19 "Control Room"
    Criterion 26 "Reactivity Control System Redundancy and Capability"
    Criterion 60 "Control of Releases of Radioactive Materials to the Environment"
    Criterion 64 "Monitoring Radioactivity Releases"

11. 10CFR20, "Standards for Protection Against Radiation"

12. NUREG - 0737, "Clarification of TMI Action Plan Requirements," November 1980, (Supplement 1, January 1983)

2.22 Personnel Radiation Dose Analysis

The Harris Nuclear Plant (HNP) radiation dose analysis has been conducted to determine the radiological impact of Steam Generator Replacement and Power Uprate Project (SGR/Uprate) which includes operation of the Model Delta 75 replacement steam generators (RSGs) at the uprated reactor core power level of 2900 MWt (NSSS power level of 2912.4 MWt.). Accident Analysis is based on 102% power, or 2958 MWt.

2.22.1 Introduction and Background

The impact of SGR/Uprate on radiation dose analysis at the HNP encompasses the following radiological dose evaluations: 1) Normal Operation Doses to the Public From Gaseous and Liquid Releases, 2) Normal Operation Doses to Onsite Personnel, 3) Design Basis Accident Doses at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ), and 4) Design Basis Accident Doses To Onsite Personnel in Vital Areas, including the Control Room.

- Normal Operation Doses to the Public from Gaseous and Liquid Releases

  Releases to the environment from normal operations after the SGR/Uprate were determined based on NUREG-0017, as implemented by the GALE Code. The offsite dose impacts from the releases were determined by using the computer programs GASPAR and LADTAP. This methodology is consistent with the HNP pre-SGR/Uprate design basis evaluations. Offsite doses continue to meet 10CFR50, Appendix I dose criteria while the effluent concentrations remain within 10CFR20 limitations.

- Normal Operation Doses to Onsite Personnel

  The reduction in reactor coolant activity, relative to previously analyzed conditions, serves to reduce in-plant radiation levels and associated personnel doses. The existing radiation zoning remains conservative for the SGR/Uprate operating conditions.

- Design Basis Accident Doses at the EAB and LPZ

  All design basis accident EAB and LPZ doses have been re-evaluated to reflect the SGR/Uprate; this includes changes to core source terms, fuel failure assumptions, and dose conversion factors. Otherwise, the basic methodology is as previously used in evaluating doses at the EAB and LPZ. There are no changes in release points or dispersion factors as a result of the SGR/Uprate. All calculated offsite doses remain within 10CFR100 limits, and within fractions of 10CFR100 limits as prescribed in NUREG-0800 for non-LOCA events.

- Design Basis Accident Doses to Onsite Personnel in Vital Areas

  All design basis accidents have been re-evaluated for the Control Room (CR), Technical Support Center (TSC), and the Emergency Operations Facility (EOF). In addition to the methodology changes discussed in the previous section, an increase in unfiltered in-leakage to 45 cfm has been assumed to provide additional operational margin. All doses remain within General Design Criterion 19 limits, and NUREG-0800 and Standard Review Plan (SRP) 6.4 guidance.

Accessibility to other Vital Areas is not significantly impacted by the SGR/Uprate changes. Post-Accident dose rates in the Reactor Auxiliary Building (RAB) were re-evaluated. The principal area requiring access is for the Post Accident Sampling System. Dose rates in this area increase between 5 and 8%, due to SGR/Uprate, as a result of consideration of fission product daughters in the Emergency Core Cooling System water. However, accessibility to this and other previously identified facilities is maintained.

Configuration Changes:

The SGR/Uprate does not affect existing radiological protection features. The analysis of the radiation dose impact on plant personnel and the general public does not involve a change to any system function credited in the FSAR for radiation mitigation and protection. The plant layout and shielding, designed to minimize personnel exposure, were not affected.

Revised Process Conditions:

Operating the plant at SGR/Uprate conditions slightly increases the generation of fission in the core and generation of activation products in the Reactor Coolant System (RCS). For accident conditions, bounding fuel failure assumptions lead to increases in accident basis RCS inventories used in many events which release RCS actuals. In addition, changes to the calculation methodology, assumptions, and operational considerations have affected the determination of onsite personnel, general public, and equipment doses for normal and accident conditions. Increased letdown flow results in an overall lowering of reactor coolant activities and offsite doses.

The results of the radiological consideration for the SGR/Uprate are presented in section 2.22.4.

2.22.2 Description of Analyses and Evaluations

Evaluations of the radiological impact of SGR/Uprate on site personnel and the general public was performed for normal and accident conditions. These evaluations are dependent upon:

- Current radioactivity source terms used in environmental determinations.
- Revised calculational methodology.
- Plant configuration (post SGR/Uprate).
- The physical nature of the sources (e.g., airborne, liquid, contained in piping, and deposited on filters).

A shielding and onsite dose evaluation was conducted, taking into account revised source terms and mass releases resulting from SGR/Uprate. It was determined that SGR/Uprate did not require any changes to HNP shielding. This review was conducted in conjunction with a review of the Reactor Coolant System (RCS) and core inventory.

Assumptions:

- Normal shielding reviews are based on 1% defective fuel. Accident shielding is based on 100% core melt source terms.

- The following inputs were changed relative to the normal offsite release and dose assumptions described in Final Safety Analysis (FSAR) Chapter 11: 1) the increase in letdown flow, 2) changes in secondary system blowdown and condensate system operations, 3) only new 10CFR20 effluent concentration (EC) limits are addressed, and 4) filtration of fuel handling building normal releases was previously erroneously credited, this has been corrected.

- The SGR/Uprate core source terms are based on a general parametric analysis of 18 month fuel cycle conditions. The analysis was performed using the ORIGEN computer code. This program is identified in the FSAR as a basis for source terms. A single set of enveloping source terms was generated, so that separate accident dose assessments are no longer required. These bounding source terms are used for onsite and offsite dose assessment.

- Fuel failure assumptions for several non-LOCA design basis accidents have been selected to envelop those which might be calculated in future reloads. Due to this set of higher, more bounding, fuel damage assumptions, the offsite doses for the following accidents, in some cases, increase: 1) Main Steam Line Break, 2) Locked Rotor, 3) Single Rod Cluster Control Assembly (RCCA) Withdrawal, 4) Misloaded Core, 5) RCCA Rejection.

- New 10CFR20 EC limits are addressed.

- Filtration of normal fuel handling building releases was previously erroneously credited, this has been corrected.

- Federal Guidance Report (FGR) 11, "Iodine Thyroid Inhalation Factors" and FGR 12, "Air Submersion, Effective Dose Equivalent Factors" are used, in accident analyses, to determine the current thyroid and equivalent-to-cloud-submersion whole body doses[1], rather than the TID-14844, Regulatory Guide (RG) 1.4 and RG 1.109. These references are an accepted alternate methodology. RG 1.109 assumptions continue to be used for 10CFR50, Appendix I evaluations.

- Iodine spikes, as described in the appropriate sections of NUREG-0800, are included in those FSAR accidents where applicable. Although not specifically required by NUREG-0800, a pre-existing iodine spike is modeled for the loss of offsite power event to provide a consistent treatment for all of the analyses.

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[1] For the Steam Generator Tube Rupture Accident, ICRP-30 Dose Conversion Factors are used. These are essentially the same as the FGR values, which were derived from ICRP-30 and supplements. Both are acceptable bases for accident dose analyses.

Operational Changes:

- Increased Letdown Flow:

  The revised Chemical and Volume Control System operation, in support of SGR/Uprate, will increase the letdown flow to include one 45 gpm orifice and one 60 gpm orifice instead of just one 60 gpm orifice as previously analyzed. The PWR-GALE computer code was used to calculate the release of radioactive material given a letdown flow rate of 106 gpm. The result of this change combined with the increase in reactor coolant volume is a decrease in normal operation reactor coolant inventory, directly resulting in a decrease in offsite doses. It also assures that spent fuel pool cooling water activity will remain within analyzed concentrations.

- Steam Generator (SG) Blowdown Processing vs. Condensate Polisher Use:

  Previous analysis assumes no treatment of the SG blowdown before it is sent to the main condenser, for cleanup by the condensate polisher. In the revised analysis the SG blowdown is demineralized, prior to being sent to the condenser. The analysis assumes that condensate polishing is not required except for chemistry control during startup. In the event of indications of primary to secondary coolant leakage, condensate resins are not regenerated, so this liquid release path will be eliminated.

2.22.3 Acceptance Criteria

Shielding for normal operations must meet the requirements of 10CFR20 related to operator dose and access control. Additional guidance for shielding is provided by USNRC Regulatory Guide 8.8 as described in FSAR Sections 12.1 and 12.3. The design of radwaste equipment must be such that the plant is capable of maintaining offsite releases and resulting doses within the requirements of 10CFR20 and 10CFR50, Appendix I. Additional guidance for evaluating compliance with these requirements is taken from USNRC Regulatory Guides 1.109 through 1.113, as discussed in FSAR Sections 11.2.3 (liquid) and 11.3.3 (gaseous). Actual performance and operation of installed equipment and reporting of actual offsite releases and doses continues to be controlled by the requirements of the Offsite Dose Calculation Manual (ODCM).

Offsite and control room doses must meet the guidelines of 10CFR100 and requirements of 10CFR50, Appendix A, General Design Criterion 19, respectively. Further acceptance criteria for specific postulated accidents are provided by the NRC in the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," NUREG-0800, which indicates each accident should be "within" (<100%), "well within" (<25%), or a "small fraction of" (≤10%) the 10CFR100 Guidelines.

Input assumption guidance for specific accidents is taken from USNRC Regulatory Guides (refer to FSAR Section 1.8 for CP&L's compliance to each RG).

1.4 Loss of Coolant Accidents
1.24 Assumptions Used for Evaluating the Potential Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure
1.25 Fuel Handling Accident
1.52 ESF Filter Systems
1.77 Control Rod Ejection
1.78 Control Room Habitability
1.109 Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents
1.111 Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors
1.112 Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors
1.113 Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I
1.143 Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants

2.22.4 Results

The plant will continue to satisfy required radiation protection requirements following the SGR/Uprate.

The results of the evaluation are broken down into the following subsections:

2.22.4.1 Shielding

The SGR/Uprate does not change system and component functions described in the FSAR. The plant layout and shielding, designed to minimize personnel exposure, were not affected. The original shielding design was based on radiation source terms developed from a reactor core thermal power of 2900 MWt (NSSS power level of 2912.4 MWt) and the equivalent of 1 percent fuel cladding defects. For SGR/Uprate, the RCS, core, and waste gas activities are based on 102 percent of the uprate core power of 2900 MWt. Significant conservatism was included in the originally calculated dose rate for shielding design. As a result, the increase in dose rate due to SGR/Uprate does not create additional inaccessible areas in the plant. Additionally, increased reactor coolant letdown processing is expected to reduce in-plant doses.

2.22.4.2 Normal Offsite Releases and Doses

The original bounding calculations prepared to evaluate conformance to 10CFR20 and 10CFR50, Appendix I demonstrate that sufficient radwaste equipment is provided in the HNP design to maintain releases within the limits of 10CFR20, Appendix B, and the resulting offsite dose to the most exposed individual within the limits of 10CFR50, Appendix I. No hardware modifications to the radwaste system are required to support SGR/Uprate.

2.22.4.3 Accident Doses

For each of the following accident analyses the major input assumptions, methodologies, and results can be found in Attachment A, with parameters applicable to many multiple events identified in Attachment A, Tables 2.22-1 and 2.22-2.

Evaluation of Radiological Consequences of Main Steam Line Break Outside Containment (FSAR 15.1.5):

The radiological consequences of the MSLB were evaluated using the assumptions of Standard Review Plan 15.1.5 and with similar methodology and parameters described in the FSAR, for the following three MSLB cases: with the following exceptions.

Description: For dose calculations the following three cases are analyzed:

1. Pre-existing iodine spike case (An SRP 15.1.5, Appendix A, Paragraph III.4.(a) Event): A reactor transient has occurred prior to the postulated MSLB and has raised the primary coolant iodine concentration to the Technical Specification 3.8.4 limit of 60 μCi/gm I-131 equivalent. The secondary coolant activity is assumed to be at the Technical Specification 3.7.1.4 limit of 0.1 μCi/gm I-131 equivalent. No fuel failure is assumed.
2. Accident Generated iodine spike case (An SRP 15.1.5, Appendix A, Paragraph III.4.(b) Event): The reactor trip and/or primary system depressurization associated with the MSLB creates an iodine spike in the primary system. The spiking model is based on an increase in the iodine release rate from the fuel rods to the primary coolant to a value that is 500 times greater that the values that yields an equilibrium reactor coolant iodine concentration of 1 uCi/gm.
3. Postulated Fuel Failure Case (An SRP 15.1.5, Appendix A, Paragraph III.5.Event): A MSLB outside containment with a bounding fuel failure assumption of 1% fuel cladding failure, and 0.7% centerline melt. This activity is released instantly to reactor coolant.

The radiological acceptance criteria are: EAB and LPZ doses less than 10% of 10CFR100 limits of 25 rem whole body (EDE) and 300 rem thyroid, for cases 1 and 2, and 100% of the limits for case 3.

Other input assumptions, methodologies and resulting doses can be found in Attachment A, Table 2.22-3.

Evaluation of Loss of AC Power to Station Auxiliaries (FSAR 15.2.6):

The evaluation of this event is performed using the same methodology as described in the FSAR. Steam Release determinations have been revised based on SGR/Uprate conditions.

Case 1: Base Case, No Iodine Spiking, which results in offsite doses that are within the 10CFR100 limits.

Case 2: Base Case, Pre-Event Iodine spike results in offsite doses less than 10CFR100 limits.

Case 3: Event Initiated Iodine Spiking results in offsite doses that are within 10CFR100 limits.

Other input assumptions, methodologies and resulting doses can be found in Attachment A, Table 2.22-4.

Evaluation of Radiological Consequences of Locked Rotor (FSAR 15.3.3):

The radiological consequences of Reactor Coolant Pump (RCP) locked rotor were evaluated assuming 8% fuel cladding failure and a 2% centerline melt with activity released to the RCS. Subsequent leakage to the steam generators and secondary side steam releases were evaluated utilizing the assumptions of the SRP. These releases result in offsite doses that are within small fraction guidelines of 10CFR100, which is consistent with the acceptance criteria.

Other input assumptions, methodologies and resulting doses can be found in Attachment A, Table 2.22-5.

Uncontrolled RCCA Withdrawal (FSAR Sections 15.4.1 and 15.4.2)

For an uncontrolled RCCA withdrawal, no fuel failure is postulated. Therefore, no offsite accident doses or onsite dose effects are anticipated. If this event were accompanied by Loss of Offsite Power and iodine spiking then the analyses in FSAR Section 15.2.6 would apply.

Evaluation of Radiological Consequences of Single RCCA Withdrawal, Core Misload (FSAR 15.4.3, 15.4.7)

The radiological consequence of a single RCCA withdrawal, and core misload were evaluated assuming 4% of the fuel cladding fails and a 2% centerline melt with activity released to the RCS. This release results in offsite doses that are a small fraction of guidelines of 10CFR100, which is consistent with the acceptance criteria.

Other input assumptions, methodologies and resulting doses can be found in Attachment A, Table 2.22-6.

Evaluation of Radiological Consequences of RCCA Ejection (FSAR Section 15.4.8)

The radiological consequences of a single RCCA rejection were evaluated assuming 4% of the fuel cladding fails and a 2% centerline melt with activity released to the RCS. This release results in offsite doses that are within 10CFR100 limits.

Other input assumptions, methodologies and resulting doses can be found in Attachment A, Table 2.22-7.

Evaluation of Radiological Consequences - Letdown Line Break Outside Containment (FSAR Section 15.6.2)

The radiological consequences of a linebreak from RCS to Outside Containment result in a small release. This release results in offsite doses that are within small fraction guidelines of 10CFR100, which is consistent with the acceptance criteria.

Other input assumptions, methodologies and resulting doses can be found in Attachment A, Table 2.22-8.

Evaluation of Radiological Consequences - Steam Generator Tube Rupture (FSAR Section 15.6.3) See NSSS LR Section 6.3.3.)

Other input assumptions, methodologies and resulting doses can be found in Attachment A, Table 2.22-9.

Evaluation of Radiological Consequences – Maximum Credible Accidnet (Loss of Coolant Accident FSAR Section 15.6.5)

The evaluation of LOCA doses uses methodologies consistent with SRP 15.6.5 guidance.

Parameter differences relative to the pre-SGR/Uprate conditions include:

1. Use of dose conversion factors from FGR 11 and 12.
2. Use of a single set of core source terms designed to envelop 18 month fuel cycle conditions.
3. A slight reduction in spray coverage from 88% to 85.9%, based on a conservative and more detailed treatment of containment volumes.
4. Use of higher unfiltered in-leakage assumptions for the CR, to provide additional operational margin.

Other input assumptions, methodologies and resulting doses can be found in Attachment A, Table 2.22-10.

Evaluation of Radioactive Waste Gas System Leak or Failure (FSAR Section 15.7.1)

This event is analyzed using RG 1.24 methodology, except for dispersion factors (per RG 1.145) and dose conversion factors (per FGR 11 and 12). The maximum quantity of radioactivity contained in each gas storage tank, based on a 1% fuel clad failure, is such that in the event of an uncontrolled release of the tank's contents, the offsite dose would be less than 500 mrem.

Other input assumptions, methodologies and resulting doses can be found in Attachment A, Table 2.22-11.

Evaluation of Liquid Waste System Leak or Failure (FSAR Section 15.7.2)

The postulated doses from this class of events have been historically small. The accident analysis has been deleted, consistent with the deletion of the SRP 15.7.2 guidance. The limiting condition is described below.

Other input assumptions, methodologies and resulting doses can be found in Attachment A, Table 2.22-12.

Evaluation of Liquid Waste Tank Failure (FSAR Section 15.7.3)

Analysis of potential releases from a failure of the refueling water storage tank, with conservative source terms assumption, indicated that ground and surface water transport pathways are such that the results are consistent with the Maximum Permissible Concentrations Limits as established in the version of 10CFR20 which was used for the original plant license.

Other input assumptions, methodologies and resulting doses can be found in Attachment A, Table 2.22-13.

Evaluation of Radiological Consequences - Fuel Handling Accident (FHA) in the Fuel Handling Building (FHB) (FSAR Section 15.7.4)

The radiological consequences of the FHA were evaluated for the uprated core inventory. Two cases were considered: an accident in the FHB and one in the containment. The accident in the FHB conforms to the guidelines of Regulatory Guide 1.25 and the Standard Review Plan. FGR 11 and 12 dose conversion factors are used. The releases in both cases result in offsite doses that are well within the 10CFR100 guidelines, which meets the acceptance criteria.

Other input assumptions, methodologies and resulting doses can be found in Attachment A, Table 2.22-14.

Evaluation of Radiological Consequences - Fuel Handling Accident in the Reactor Containment Building (FSAR Section 15.7.4)

The parameters used in the existing analysis were evaluated to determine the impact of SGR/Uprate core source terms and the use of FGR 11 and 12 source terms. This evaluation indicates that doses would be reduced by a factor of 0.745 for Thyroid, 0.474 for Whole Body, and 0.902 for β-Skin. For simplicity and conservatism, the pre-SGR/Uprate doses are retained.

Other input assumptions, methodologies and resulting doses can be found in Attachment A, Table 2.22-14.

Cask Drop Accident (FSAR Section 15.7.5)

The current accident analysis is based on Fuel Transferred from other CP&L stations, and is not impacted by the SGR/Uprate conditions.

Other input assumptions, methodologies and resulting doses can be found in Attachment A, Table 2.22-15.

2.22.5 Conclusions

The existing plant design, radiation protection measures, procedures and operating practices combine to keep onsite and general public exposures within regulatory limits and industry guidelines in accordance with the FSAR.

No changes or additions to structures, equipment, or procedures are necessary to provide adequate radiation protection for the operators or the public during normal or post-accident operations to support the SGR/Uprate. The existing structures, systems, and components can safely handle the changes in post accident source terms and releases from the SGR/Uprate conditions, and resulting onsite and offsite doses are less than the 10CFR guidelines and are within the SRP recommendations. Therefore, the radiological consequence acceptance criteria for postulated Condition II, III, and IV events are satisfied. These results are consistent with the current design and licensing bases discussed in the FSAR.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS thermal power level of 2787.4 MWt.

2.22.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.4.8 Specific Activity
   Table 3.3-6 Radiation Monitoring Instrumentation for Plant Operations
   6.11 Radiation Protection Program
   6.12 High Radiation Area

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant (Units 1 and 2)," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"

9. 10CFR20, "Standards For Protection Against Radiation"

10. 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants"
    Criterion 19, "Control Room"
    Criterion 60" "Control of Radioactive Releases to the Environment"
    Criterion 64, "Monitoring Radioactive Releases"

11. 10CFR50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents"

12. 10CFR100, "Reactor Site Criteria"

13. RG 8.8, " Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," (Rev. 3, 6/78)

14. RG 1.4, "Assumptions Used For Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident," (Rev.2, 6 /74)

15. RG 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," (Rev. 1, 6/74)

16. RG 1.24, "Assumptions Used for Evaluating the Potential Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure," (Rev. 0, 3/72)

17. RG 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences Of A Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," (Rev. 0)

18. RG 1.49, "Power Levels of Nuclear Power Plants," (Rev. 1, 12/73)

19. RG 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants," (Rev. 2, 3/78)

20. RG 1.77, "Assumptions Used for Evaluating A Control Rod Ejection Accident for Pressure and Water Reactors," (Rev. 0, 5/74)

21. RG 1.78, "Assumptions for Evaluating the Habitability Of A Nuclear Power Plant Control Room During A Postulated Hazardous Chemical Release," (Rev. 0, 6/74)

22. RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," (Rev.1, 10/77)

23. RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," (Rev. 1, 7/77)

24. RG 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," (Rev. 0-R, 5/77)

25. RG 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," (Rev. 1, 4/77)

26. RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," (Rev. 1, 10/79)

27. 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, dated: September 1988

28. Federal Guidance Report No. 12, External Exposure to Radionuclides in Air, Water, and Soil, EPA 402-R-93-81, dated: September 1993

29. Branch Technical Position ESTB 11-5, "Postulated Radioactive Releases due to a Waste Gas System Leak or Failure"

30. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," 1962, USAEC

31. Murphy, K.G., and Campe, K.M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," 13th AEC Air Cleaning Conference, August 1974

32. CQL-00-195, "Final Engineering Report Section 6.3.3 - Radiological Consequences Analysis (WX1520)"

33. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988

Attachment A – Major Input Assumptions, Methodologies, and Results of the Accident Analyses

Dose consequences for all design basis accidents and events previously addressed in the FSAR were re-evaluated for the SGR/Uprate. The major input assumptions, methodologies, and results of these accident analyses provided in Tables 2.22-1 through 2.22-15. These accident analyses reflect the impact of the SGR/Uprate. They also reflect certain other accident analysis assumption and methodology changes that are also identified in the Tables.

Table	FSAR Section	Accident or Event
2.22-1	General	General Accident Analysis Parameters: Dispersion Factors, Dose Conversion Factors, and Control Room, TSC, and EOF Ventilation System Parameters
2.22-2	Source Terms	Iodine And Noble Gas Inventory In Reactor Core, Gap And Coolant, For Use In Design Basis Accident Analyses
2.22-3	15.1.5	Steam System Piping Failure [Limiting Event: Main Steam Line Break Outside Containment]
2.22-4	15.2.6	Loss of Non-Emergency AC Power to the Station Auxiliaries
2.22-5	15.3.3, 15.3.4	Reactor Coolant Pump Shaft Seizure (Locked Rotor) Reactor Coolant Pump Shaft Break [Limiting and Analyzed Event: Locked Rotor]
2.22-6	15.4.3 15.4.7	Single RCCA Withdrawal Misloaded Core [Same Enveloping Fuel Damage Postulated]
2.22-7	15.4.8	Spectrum of Rod Cluster Control Assembly (RCCA) Ejection Accidents [Limiting Event: RCCA Ejection]
2.22-8	15.6.2	Break in Instrument Line or Other Line from Reactor Coolant Pressure Boundary that Penetrate Containment [Limiting Event: Letdown Line Break outside Containment]
2.22-9	15.6.3	Steam Generator Tube Rupture
2.22-10	15.6.5	Loss of Coolant Accident (LOCA) [Limiting Event: Large Break LOCA based Maximum Credible Accident]
2.22-11	15.7.1	Radioactive Waste Gas System Leak or Failure
2.22-12	15.7.2	Liquid Waste System Leak or Failure
2.22-13	15.7.3	Postulated Radioactive Release Due to Liquid Tank Failure
2.22-14	15.7.4	Fuel Handling Accident
2.22-15	15.7.5	Cask Drop Accident

TABLE 2.22-1 GENERAL ACCIDENT ANALYSIS PARAMETERS	
Parameter	**Value**
Dispersion Factors (No change from those currently used for design basis accident analyses as described in the FSAR.)	**Offsite Dose values:** 0 – 2 hr EAB 6.17E-4 sec/m**3 0 – 8 hr LPZ 1.40E-4 sec/m**3 8 –24 hr LPZ 1.00E-4 sec/m**3 1 – 3 day LPZ 5.90E-5 sec/m**3 4 –30 day LPZ 2.40E-5 sec/m**3 **Control Room values:** X/Q(0- 8hr@CR) =4.08E-3 sec/m**3 (8-24hr@CR) =1.16E-3 sec/m**3 (1- 4day@CR)=3.25E-4 sec/m**3 (4-30day@CR)=1.23E-5 sec/m**3
Breathing Rates (Per R.G. 1.4, TID-14844, and SRP 6.4 guidance) (Ref 14, 30, 8)	**OFFSITE** 0 - 8Hr: 3.47E-4 m**3/sec 8 - 24Hr: 1.75E-4 m**3/sec 24-720Hr: 2.32E-4 m**3/sec **ONSITE** For the CR, TSC, and EOF the breathing rate is constant at 3.47E-4 m**3/sec.
Occupancy Factors (Per SRP 6.4, Ref 8)	The control room occupancy factors, also used for the TSC and EOF: 0- 1 day = 1.0 1- 4 day = 0.6 4-30 day = 0.4
Whole Body Dose Conversion Factors Values from Federal Guidance Report 12 (Ref 28), Table III.1: Dose Coefficients for Air Submersion. EDE values are used to provide Whole Body Dose risk-equivalent results.	W.B. DCF's to be used are: KR-83m 5.55E-06 REM-m**3/sec-Ci Kr-85 4.40E-04 Kr-85m 2.77E-02 Kr-87 1.52E-01 Kr-88 3.77E-01 Xe-131m 1.44E-03 Xe-133m 5.07E-03 Xe-133 5.77E-03 Xe-135m 7.55E-02 Xe-135 4.40E-02 Xe-138 2.13E-01
Iodine (Thyroid) Dose Conversion Factors Federal Guidance Report 11 (Ref 27) Table 2.1	Isotope DCF (REM/Ci inhaled) I-131 1.08E+06 I-132 6.44E+03 I-133 1.80E+05 I-134 1.07E+03 I-135 3.13E+04

TABLE 2.22-1 GENERAL ACCIDENT ANALYSIS PARAMETERS	
Parameter	**Value**
Beta-skin Dose Conversion Factors Values from Reg. Guide 1.109 (Ref 22), converted to the stated units by multiplying by 31.7.	Beta-skin DCF's to be used are: KR-83m 0.00E-00 REM-m**3/sec-Ci Kr-85 4.25E-02 Kr-85m 4.63E-02 Kr-87 3.08E-01 Kr-88 7.51E-02 Xe-131m 1.51E-02 Xe-133m 3.15E-02 Xe-133 9.70E-03 Xe-135m 2.25E-02 Xe-135 5.90E-02 Xe-138 1.31E-01
Ventilation Flow Rates, Volumes, Filter Eff., and IPF for Control Room Note that inhalation protection may be assessed using the CONTROOM program, which assesses time dependent concentrations, or by the simpler Iodine Protection Factor (IPF) from Murphy-Campe. Geometry Factor (GF) is also based on Murphy-Campe (Ref 31), Eq. 9.	Control Room Parameter — CFM Unfiltered in-leakage (cfm): 45 Filtered Intake (cfm) 400 Recirculation Flow (cfm) 3600 Volume (cu. ft.) 80,180 Filter Efficiency 99% Geometry Factor 25.8 Iodine Protection Factor 51.1
Fuel Damage Release Fraction to Coolant Parameters for non-LOCA:	
Gap Activity Fraction of Core Activity:	Noble Gases except Kr-85: 0.10 (per R.G. 1.25) Kr-85 0.30 (per R.G. 1.25) Iodines Except I-131 0.10 (per R.G. 1.25, Ref 17) I-131 0.12 (per NUREG/CR-5009, Ref 33)
Centerline Melt Fractions of Core Activity Released to Coolant:	Noble Gases 1.0 Iodines 0.5
Core Thermal Power	All accident analyses are based 2958 MWt (102%) Some accidents are based on current Technical Specification limits. Examples include I-131 equivalent reactor and secondary coolant concentrations, and waste gas decay tank inventory limits.

TABLE 2.22-2					
IODINE AND NOBLE GAS INVENTORY IN REACTOR CORE, GAP AND COOLANT, FOR USE IN DESIGN BASIS ACCIDENT ANALYSES					
Isotope	Decay Constant (1/sec)	Core Activity (Ci)	Gap Fractions	Gap Activity (Ci)	Reactor Coolant* (uCi/g)
I-131	9.98E-07	8.04E+07	0.12	9.65E+06	1.71E+00
I-132	8.37E-05	1.16E+08	0.1	1.16E+07	2.47E+00
I-133	9.26E-06	1.64E+08	0.1	1.64E+07	7.23E+00
I-134	2.20E-04	1.80E+08	0.1	1.80E+07	5.67E-01
I-135	2.91E-05	1.53E+08	0.1	1.53E+07	1.84E+00
Kr-83m	1.05E-04	1.02E+07	0.1	1.02E+06	4.17E-01
Kr-85m	4.30E-05	2.19E+07	0.1	2.19E+06	1.73E+00
Kr-85	2.05E-09	8.65E+05	0.3	2.60E+05	1.06E+01
Kr-87	1.51E-04	4.22E+07	0.1	4.22E+06	1.10E+00
Kr-88	6.78E-05	5.95E+07	0.1	5.95E+06	3.21E+00
Xe-131m	6.74E-07	8.99E+05	0.1	8.99E+04	3.41E+00
Xe-133m	3.67E-06	5.15E+06	0.1	5.15E+05	4.86E+00
Xe-133	1.53E-06	1.62E+08	0.1	1.62E+07	2.76E+02
Xe-135m	7.56E-04	3.21E+07	0.1	3.21E+06	4.36E-01
Xe-135	2.12E-05	3.84E+07	0.1	3.84E+06	8.52E+00
Xe-138	8.15E-04	1.38E+08	0.1	1.38E+07	6.30E-01
* Conservatively, based on 1% cladding defects					

TABLE 2.22-3 Steam System Piping Failure (FSAR 15.1.5) [Limiting Event: Main Steam Line Break Outside Containment]	
Core Thermal Power (MWt)	2958
Offsite Power	Lost
Source Terms	1.Pre-existing iodine spike case (An SRP 15.1.5, Appendix A, Paragraph III.4.(a) Event): A reactor transient has occurred prior to the postulated MSLB and has raised the primary coolant iodine concentration to the Technical Specification 3.8.4 limit of 60 μCi/gm I-131 equivalent. The secondary coolant activity is assumed to be at the Technical Specification 3.7.1.4 limit of 0.1 μCi/gm I-131 equivalent. No fuel failure is assumed. 2. Accident Generated iodine spike case (An SRP 15.1.5, Appendix A, Paragraph III.4.(b) Event): The reactor trip and/or primary system depressurization associated with the MSLB creates an iodine spike in the primary system. The spiking model is based on an increase in the iodine release rate from the fuel rods to the primary coolant to a value that is 500 times greater that the values that yields an equilibrium reactor coolant iodine concentration of 1 uCi/gm. 3. Postulated Fuel Failure Case (An SRP 15.1.5, Appendix A, Paragraph III.5.Event): A MSLB outside containment with a bounding fuel failure assumption of 1% fuel cladding failure, and 0.7% centerline melt. This activity is released instantly to reactor coolant
Primary to secondary leak rate	0.3 gpm from affected SG 0.7 gpm from un-affected SGs
Iodine Partition Factor in Steam Generators	1 for the faulted Steam Generators (assumed un-isolated) 0.01 for the intact Steam Generators
Duration of plant cooldown by secondary system after accident	8 hours (Note that HNP T.S. 3.4..2 limit leakage through any 1 steam generator to 150 gpd which is 0.104 gpm. Therefore, the above values are a significant conservatism. At 8 hours, the reactor cooling is assumed to be by the RHR system, and cooling using the steam generators and the atmospheric dump valve is assumed to have ceased. Thus, releases from the not-affected steam generators have ceased. Some release from the affected steam generator could continue if the isolation valve is not closed. However, given the above conservatism in primary to secondary leakage rate treatment it is considered acceptable to cease the radiological evaluation at 8 hours for this pathway as well. The assumed leakage is almost three times the expected value for the first 8 hours. Improved X/Qs would also apply during the 8-24 hour period. Therefore, the existing 8 hour analysis is bounding.
Initial steam release from defective steam generator	162,000 lbs

Steam release from two non-defective steam generators	386,000 lbs (0-2 hrs) 892,000 lbs (2-8 hrs)
Feedwater flow to two non-defective steam generators	482,000 lbs (0-2 hrs) 967,000 lbs (2-8 hrs)
Resulting Doses	See Below

CASE 1: Pre-Existing Iodine Spike, no Fuel Damage

SRP 15.1.5, Appendix A, Section III.4.(a) based Event

EAB	LPZ	CR	
3.67E+00	2.17E+00	7.72E-01	Total Thyroid Dose (rem)
1.01E-03	9.14E-04	1.03E-03	Total Whole Body Dose (rem)
		3.40E-02	Total β-Skin Dose (rem)

CASE 2: Accident Generated Iodine Spike, no Fuel Damage

SRP 15.1.5, Appendix A, Section III.4.(b) based Event

EAB	LPZ	CR	
5.38E+00	1.31E+01	4.66E+00	Total Thyroid Dose (rem)
1.01E-03	9.14E-04	1.03E-03	Total Whole Body Dose (rem)
		3.40E-02	Total β-Skin Dose (rem)

CASE 3: Bounding Fuel Damage

SRP 15.1.5, Appendix A, Section III.5.based Event

EAB	LPZ	CR	
8.45E+01	7.57E+01	2.70E+01	Total Thyroid Doses (rem)
7.03E-01	6.38E-01	7.20E-01	Total Whole Body Doses (rem)
		2.02E+01	Total β-Skin Dose (rem)

The offsite doses for case 1 and 2 are within a small fraction (10%) of 10CFR100 (Ref 12) limits. Case 3 doses are within 10CFR100 limits. Onsite doses are within GDC-19 and SRP 6.4 (Ref 8) limits.

TABLE 2.22-4 **Loss of Non-Emergency AC Power to the Station Auxiliaries (FSAR 15.2.6)**	
Core Thermal Power (MWt)	2958
Source Terms	1. A base case with no iodine spike case: The reactor coolant is at the Technical Specification Limit of 1 μCi/gm I-131 eq. The secondary coolant activity is assumed to be at the Technical Specification limit of 0.1 μCi/gm I-131 eq. 2. A Pre-Event iodine spike case: The reactor coolant is at the Technical Specification Limit of 60μCi/gm I-131 eq. The secondary coolant activity is assumed to be at the Technical Specification limit of 0.1 μCi/gm I-131 eq. 3. An Event Initiated Iodine Spiking Case: The reactor coolant experiences an immediate increase in Iodine appearance rate by a Factor of 500. For simplicity and conservatism, this rate is assumed to reach equilibrium instantly, resulting in a 500 μCi/gm I-131 eq. concentration, for the duration of the event. No credit is taken for depletion of available gap activity.
Primary to secondary leak rate	1 gpm
Iodine Partition Factor in Steam Generators	0.01
Duration of plant cooldown by secondary system after accident	8 hours
Steam release from steam generators	364,000 lbs (0-2 hrs) 939,000 lbs (2-8 hrs)
Feedwater flow to steam generators	508,000 lbs (0-2 hrs) 1,052,000 lbs (2-8 hrs)
Resulting Doses	See Below

Only the limiting Thyroid Doses are calculated. The resulting doses (rem thyroid) are:

CASE	EAB	LPZ	CR
1	3.90E-02	3.18E-02	1.13E-02
2	8.33E-02	5.40E-02	1.92E-02
3	4.14E-01	3.59E-01	1.28E-01

TABLE 2.22-5 **Reactor Coolant Pump Locked Rotor (FSAR 15.3.3)**	
Core Thermal Power (MWt)	2958
Source Terms	Postulated Fuel Failure: A bounding fuel failure assumption of 8% fuel cladding failure, and 2% centerline melt. This activity is release instantly to reactor coolant
Primary to secondary leak rate	1 gpm
Iodine Partition Factor in Steam Generators	0.01
Duration of plant cooldown by secondary system after accident	8 hours
Steam release from steam generators	364,000 lbs (0-2 hrs) 939,000 lbs (2-8 hrs)
Feedwater flow to steam generators	508,000 lbs (0-2 hrs) 1,052,000 lbs (2-8 hrs)
Resulting Doses	See Below

CASE	EAB	LPZ	CR
Thyroid (rem)	8.27E+00	8.67E+00	3.09E+00
W.B. (rem)	1.79E+00	8.56E-01	9.66E-01
Skin (rem)			2.10E+01

The offsite doses are within a small fraction (10%) of 10CFR100 (Ref 12) limits. Onsite doses are within GDC-19 and SRP 6.4 (Ref 8) limits.

TABLE 2.22-6 Single RCCA Withdrawal (FSAR 15.4.3) Misloaded Core (FSAR 15.4.7)	
Core Thermal Power (MWt)	2958
Source Terms	Postulated Fuel Failure: A bounding fuel failure assumption of 4% fuel cladding failure, and 2% centerline melt. This activity is released instantly to reactor coolant
Primary to secondary leak rate	1 gpm
Iodine Partition Factor in Steam Generators	0.01
Duration of plant cooldown by secondary system after accident	8 hours
Steam release from steam generators	364,000 lbs (0-2 hrs) 939,000 lbs (2-8 hrs)
Feedwater flow to steam generators	508,000 lbs (0-2 hrs) 1,052,000 lbs (2-8 hrs)
Resulting Doses	See Below

CASE	EAB	LPZ	CR
Thyroid (rem)	6.18E+00	6.48E+00	2.31E+00
W.B. (rem)	1.52E+00	7.24E-01	8.18E-01
Skin (rem)			1.77E+01

The offsite doses are within a small fraction (10%) of 10CFR100 (Ref 12) limits. Onsite doses are within GDC-19 and SRP 6.4 (Ref 8) limits.

TABLE 2.22-7 RCCA Ejection Accident (FSAR 15.4.8)	
Core Thermal Power (MWt)	2958
Source Terms	Postulated Fuel Failure: A bounding fuel failure assumption of 4% fuel cladding failure, and 2% centerline melt. This activity is released instantly to reactor coolant
Release Paths Two Independent Release Paths are evaluated. No credit is taken for primary system depressurization by the RCCA caused LOCA in evaluating doses from the primary to secondary leakage pathway.	1. Release of fission products through primary to secondary leakage that is subsequently released via a steam dump. 2. A loss of coolant through the break (LOCA) to the primary containment that is subsequently leaked through the primary containment.
Primary to secondary leak rate	1 gpm
Iodine Partition Factor in Steam Generators	0.01
Duration of plant cooldown by secondary system after accident	8 hours
Steam release from steam generators	364,000 lbs (0-2 hrs) 939,000 lbs (2-8 hrs)
Feedwater flow to steam generators	508,000 lbs (0-2 hrs) 1,052,000 lbs (2-8 hrs)
Primary Containment Release Path Credited Removal Mechanisms	Same and design basis LOCA, i.e. 50% initial plateout, containment spray removal credit.
Resulting Doses	See Below

PRIMARY to SECONDARY RELEASE PATHWAY

CASE	EAB	LPZ	CR
Thyroid (rem)	6.18E+00	6.48E+00	2.31E+00
W.B. (rem)	1.52E+00	7.24E-01	8.18E-01
Skin (rem)			1.77E+01

RELEASE TO CONTAINMENT PATHWAY

CASE	EAB	LPZ	CR
Thyroid (rem)	1.68E+00	2.17E+00	3.49E-01
W.B. (rem)	4.03E-02	2.44E-02	1.20E-02
Skin (rem)			2.49E-01

These offsite doses are well within (<25%) of 10CFR100 (Ref 12) limits. Onsite doses are within GDC-19 and SRP 6.4 (Ref 8) limits.

TABLE 2.22-8 **Letdown Line Break Outside Containment (FSAR 15.6.2)**	
Core Thermal Power (MWt)	2958
Source Terms	Design Basis Reactor Coolant Activity
Assumed Letdown Orifices Open	Expected 2, including 45gpm and 60 gpm orifices Assumed 2, including both 60 gpm orifices
Break Flow	Conservatively, 200 gpm
Duration before break is manually isolated.	30 minutes
Resulting Doses	See Below

(Rem)	EAB	LPZ	CR
Thyroid	6.26E+00	1.42E+00	5.06E-01
Whole Body	5.03E-02	1.14E-02	1.29E-02
Skin			4.24E-01

The offsite doses are substantially below (a small fraction, <10%) of 10CFR100 (Ref 12) limits. Onsite doses are within GDC-19 and SRP 6.4 (Ref 8) limits.

TABLE 2.22-9 **Steam Generator Tube Rupture (FSAR 15.6.3)**
Summary of parameters and results are contained in NSSS Licensing Report Section 6.3.3 – Radiological Consequences Analysis [for SGTR]

These offsite doses are well within (<25%) of 10CFR100 (Ref 12) limits. Onsite doses are within GDC-19 and SRP 6.4 (Ref 8) limits.

TABLE 2.22-10 **Large Break LOCA based Maximum Credible Accident (FSAR 15.6.5)**	
Core Thermal Power (MWt)	2958
Source Terms	100% of Core Noble Gases, and 50% of Core Iodines Design Basis Reactor Coolant Activity Initially Released to Containment
Plateout	50% of Iodines at Time 0
Iodine Form	91% Elemental 4% Organic 5% Particulate
Containment Leak Rate	0.1%/day for first 24 hours 0.05%/day from 1 to 30 days
Containment Spray System Parameters	
Containment Volume	2.344E6 cu. ft. (maximum with uncertainty eval.)
Start of Containment Spray	120 seconds
Spray Coverage	85.9%
Spray Flow (single Train)	1730 gpm
Liquid Volume in Sump	38,500 cu. ft.
Effective Iodine Partition Coefficient	5000
Fall height	125 ft.
Elemental Iodine Spray Lambda	20 / hr
Particulate Iodine Spray Lambda	3.94 /hr until DF of 50, 0.394 thereafter.
Sprayed/Unsprayed Region Exchange Rate	1.60
ECCS Leakage Parameters	
Leakage	0.967 gpm Inside RABEES (filtered release) 0.033 gpm Outside RABEES (unfiltered release)
ECCS Water Volume	360,000 gallons
Earliest Start of Recirc	20 minutes
Radioactivity in ECCS Liquid	50% of Core Iodines
Flashing Fraction	2%, based on sump water temperature
RABEES Filter Efficiency	95%
Resulting Doses	See Below

(Rem)	EAB	LPZ	CR
Thyroid	86.8	172	28.0
Whole Body	1.88	1.23	0.547
Skin			11.3

The offsite doses are within 10CFR100 (Ref 12) limits. Onsite doses are within GDC-19 and SRP 6.4 (Ref 8) limits.

TABLE 2.22-11 **Radioactive Waste Gas System Leak or Failure (FSAR 15.7.1)**	
Core Thermal Power	2958 MWt
Fuel Cladding Defect Analysis Basis	1%
Limiting Event	Failure of a single Gaseous Waste Decay Tank (GWDT). Inventory based on R.G. 1.24 based assessment of maximum activity associated with post-shutdown degassing, with 1% fuel cladding defects. This inventory is conservative relative to a GWDT maximum operating inventory with 1% fuel cladding defects, as would be determined based on BTP ETSB 11-5.
Gas Decay Tank Inventory (Curies)	Kr-83m 19.1 Kr-85m 138 Kr-85 4,100 Kr-87 46.0 Kr-88 172 Xe-131m 775 Xe-133m 903 Xe-133 58,500 Xe-135m 56.6 Xe-135 900 Xe-138 5.16
Calculated Offsite Doses with Above Source	0.29 rem W.B. EAB 0.065 rem W.B. LPZ
Technical Specification Limit	T.S. 6.8.4.j requires GWDT inventories to be such that doses would be less than 0.5 rem W.B. per ETSB 11-5 methodology. Design basis accident analysis inventories would be within this specification, even if, as is currently done, R.G. 1.109 dose conversion factors were used to establish compliance.

TABLE 2.22-12 **Liquid Waste System Leak or Failure (FSAR 15.7.2)**
This accident analysis has been deleted based on the fact that the limiting Liquid Radioactive Water System Failure is now addressed in FSAR 15.7.3, and 2.4.12.

TABLE 2.22-13 **Liquid Tank Failure (FSAR 15.7.3)**
This accident analysis is impacted slightly by changes in the assumed Refueling Water Storage Tank (RWST) inventory. Calculated offsite concentrations from surface water and groundwater pathways continue to be within the design basis old 10CFR20 maximum permissible concentration limits

TABLE 2.22-14
Fuel Handling Accidents (FSAR 15.7.4)

FHA in Fuel Handling Building

Fuel Damage	264 rods in dropped assembly, and 50 rods in struck assembly. 314 rods total.
Radial Power Peaking Factor	1.73
Overall Iodine DF	72, based on 22 foot water depth
FHB Exhaust Filter Efficiency	95%
Resulting Doses	See Below

CASE	EAB	LPZ	CR
Thyroid (rem)	1.44E+01	3.28E+00	1.16E+00
W.B. (rem)	4.47E-01	1.01E-01	1.15E-01
Skin (rem)			5.73E+00

FHA in Containment

Existing Analyses Not Re-performed, however impacts of changes is core source terms, and dose conversion factors evaluated to provide dose adjustment factors. All adjustment factors are less than 1, so doses are not increased. The pre-SGR/Uprate accident analyses for this event is bounding for the SGR/Uprate condition.

These offsite doses are well within (<25%) of 10CFR100 (Ref 12) limits. Onsite doses are within GDC-19 and SRP 6.4 (Ref 8) limits.

TABLE 2.22-15
Cask Drop Accident (FSAR Section 15.7.5)

The current accident analysis is based on Fuel Transferred from other CP&L Stations, and is not impacted by Harris Station Power Uprate or Steam Generator Replacement

2.23 Equipment Qualification

The assessment of electrical and mechanical safety-related equipment qualification (EQ) for the Harris Nuclear Plant (HNP) has been conducted to determine the impact of the Steam Generator Replacement and Power Uprate (SGR/Uprate). This project includes operation of the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.23.1 Introduction and Background

As stated in the Final Safety Analysis Report (FSAR), equipment that is relied on to perform a necessary safety function must be demonstrated capable of maintaining functional operability under all service conditions (including exposure to radiation) postulated to occur for the duration it is required to operate, during its installed life[1]. Existing electrical and mechanical equipment qualification documentation packages (EQDPs) provide equipment-specific documented evidence which confirms the required safety-related functional equipment performance under current (pre-SGR/Uprate) normal and accident conditions.

FSAR Section 3.11 provides information on the environmental conditions and design bases for which safety-related electrical and mechanical equipment is designed to ensure compliance with 10CRF50.49. It consists of written descriptions, tables, figures, appendices, and data references which delineate the requirements and the demonstrated qualifycations of safety-related plant equipment. Plant zone maps summarize normal and postulated accident equipment exposures in terms of peak radiation and temperature/ pressure/humidity conditions.

<u>Configuration Changes</u>:

The analysis of the radiation dose impact on EQ in the radiation zones does not involve a change to any system function credited in the FSAR for radiation mitigation and protection. The analysis of environmental design and qualification of safety-related equipment for SGR/Uprate does not involve a physical change to any plant safety-related system. The plant layout and shielding (designed to minimize equipment exposure to harsh environmental effects) were not changed by the SGR/Uprate. SGR/Uprate does not adversely impact the environmental design of electrical and mechanical equipment and does not involve a change to environmentally qualified safety-related equipment.

<u>Revised Process Conditions</u>:

Evaluation of HNP equipment qualifications under SGR/Uprate conditions considered the following changes in equipment environmental conditions:

- Increased exposure to total integrated radiation doses (TIDs);

[1] In the case of radiation exposure, the installed life assumes a normal, 40 year, operational exposure plus an additional one-year post-accident exposure.

- Revised accident temperature and pressure profiles inside the containment and main steam line tunnel (per mass and energy [M&E] releases defined in BOP LR Section 2.24, "Containment and Subcompartment Analysis"); and
- Changes in maximum process temperatures (for impact to current mechanical equipment qualifications).

2.23.2 Description of Analyses and Evaluations

EQ was evaluated to ensure that the above-noted SGR/Uprate changes to normal and/or postulated accident conditions do not adversely affect the capability of safety-related equipment to perform their intended safety functions, in accordance with the FSAR. This evaluation was completed by: comparing current EQ requirements (defined within existing EQDPs) to SGR/Uprate EQ requirements (summarized herein); and identifying increases for their impact to the existing documentation of equipment configuration and performance.

Changes to the radiation dose calculation methodology, assumptions, and operational considerations have also affected the determination of EQ radiation doses for normal and accident conditions. Furthermore, the accident dose analyses were performed at 102% of uprate power to compensate for instrument uncertainty as recommended in Regulatory Guide 1.49. The following factors were considered relative to increased radiation affects:

a) The methodology used for evaluating EQ radiation dose contribution utilized the Radiation Zones as shown in FSAR Section 3.11B. The Containment, Reactor Auxiliary Building (RAB) and Fuel Handling Building (FHB) are divided into Radiation Zones with each zone considering:

   - 40-year normal operation dose
   - 1-day integrated accident dose
   - 1-month integrated accident dose
   - 1-year integrated accident dose

b) All safety-related equipment in a zone is subject to the normal 40-year dose, but only equipment that is required to be available to perform post-accident functions are qualified with consideration given the time constraints on the required functions.

c) The normal 40-year operation dose is dependent primarily on the N-16 activity inside the Containment and the primary coolant concentrations for areas inside the RAB, with the exception of the Demineralizers. The doses for the Demineralizers reflect the primary coolant activity removed from the process stream as well as operational considerations (e.g., changing the resin more frequently). Other EQ dose analysis considerations include:

   - Containment Core and Maximum Credible Accident (MCA) Airborne Gamma Activity
   - Containment MCA Airborne Beta Activity
   - Containment MCA Plateout Activity

- Containment MCA Sump Activity
- Containment Normal Operation (N-16 driven) Activity
- Reactor Auxiliary Building (RAB) Airborne for MCA
- RAB Filters (Control Room and Engineered Safety Features) for MCA
- RAB Piping for MCA
- RAB Normal (RCS activity based)
- RAB Normal (Demineralizer based)
- Fuel Handling Building (FHB) for Fuel Handling Accident (various factors)
- FHB for Normal Operation (RCS activity based)

d) The evaluation reflects the uprated MCA source terms. Normal operation uprate factors are chiefly dependent on design basis RCS activity changes.

Acceptance criteria, relevant to the equipment qualification evaluation, are identified in Section 2.23.3. The results of the equipment qualification evaluation to satisfy applicable design and licensing bases are presented in Section 2.23.4.

2.23.2.1 Radiation Dose Evaluations

To determine the radiation impact on equipment qualification, due to SGR/Uprate, revised operational and calculation assumptions were considered to reflect industry practices, current regulatory requirements and associated guidance. Many of these factors are dependent on SGR/Uprate, such as core source terms.

The evaluation takes into account the impact of airborne activity due to daughter products. Additionally, this method of evaluating the effects of daughter products to the equipment environment also considers plateout on the primary containment walls and collection/accumulation in the containment sump. The maximum change in radiation level, considering the combination of activity of daughter product plateout and sump accumulation, along with SGR/Uprate, will increase 11% for inside containment and 8% for outside containment.

The radiation margins presently documented in the EQDPs are substantially higher than the required 10% margin per IEEE Standard 323-1974. The radiation margins now documented in the EQDPs would somewhat decrease, however not below the required 10% margin. Therefore the increased radiation of the SGR/Uprate condition will have no adverse impact on equipment qualification.

2.23.2.2 Accident Temperature/Pressure Qualification Evaluation

Normal temperature and pressure profiles used in the environmental qualification of safety-related equipment are not adversely affected by the SGR/Uprate. In addition, the SGR/Uprate does not adversely affect the existing environmental qualification of safety-related equipment with regard to chemical exposure, and spray. Changes in submergence level do not expose safety-related equipment to submergence effects.

Accident temperature and pressure profiles were evaluated for equipment inside the containment and the main steam line tunnel (MSLT). Evaluation of containment temperature and pressure conditions utilized composite LOCA/MSLB profiles. Accident environmental parameters for containment and MSLT are summarized in Table 2.23-1 and Table 2.23-2, respectively. Furthermore, LR Figures 2.23-1 through 2.23-4 (as noted in these LR Tables) provide time-related plots of containment and MSLT conditions.

For containment temperature conditions, the SGR/Uprate peak accident temperature is less than the current peak value. Similarly, for ease of comparison and presentation, all three SGR/Uprate average temperatures (over the initial 24-hour transient and subsequent 24-hour to 365-day post-accident period, as well as over the entire 365-day accident duration) are less than the corresponding current average values. Therefore, the accident condition temperature margins in the existing EQDPs will improve slightly for post-SGR/Uprate postulated conditions. This improvement was confirmed based upon a post-accident degradation equivalency calculation (using NRC-endorsed Arrhenius techniques), which has been summarized per the following discussion:

- LR Figure 2.23-5 overlays the current HNP FSAR Figure 3.11.4-1 composite accident temperature profile and the SGR/Uprate composite profile shown in LR Figure 2.23-1.
- With an objective to evaluate a typical 'like-for-like' time/temperature-dependent rate of degradation, the degradation associated with the current HNP (FSAR) composite accident temperature profile was compared against the SGR/ Uprate composite profile shown in LR Figure 2.23-1.
- A 'constant reference' of 120°F with a 1.0 eV material activation energy was used for these comparisons. Alternate reference temperatures and/or activation energies should yield similar results for comparison/evaluation purposes; 120°F was selected to assure that all portions of both composite profiles were included within the comparison, while 1.0 eV was more arbitrarily chosen (given its conservative, but typical, value).
- CP&L-licensed "System 1000" program (previously supplied by Fulcrum Group, and currently maintained by RCM Technologies [Huntsville, AL]) was utilized to produce the post-accident degradation equivalency calculation. The benchmark verification results were successfully produced for this QA-controlled software program prior to the calculation of degradation equivalency results.
- Based upon the software's program interface/terminology (owing to its normal application), the program is designed to generate equivalency comparisons of test report profiles ["accident test"] in relation to a plant-specific qualification requirements profile ["accident requirements"]. The normal program application confirms that test conditions exceed plant requirements, by 'overlapping' their time/temperature histories and computing the differences in their thermal degradations over time. For the purpose of this evaluation, "accident requirements" were selected to be the SGR/Uprate composite profile and "accident test" parameters were based upon the current composite profile. With this selection convention for purposes of this specific comparison, the acceptance criterion still requires that positive margin exist for this degra dation equivalency comparison process. Positive margin confirms that the current profile envelops (or exceeds) the SGR/Uprate profile.

- The current composite profile exceeded the SGR/Uprate composite profile by positive minimum margin of 0.275 – Years or 11.35%. Therefore, the SGR/ Uprate composite profile [LR Figure 2.23-1], taken over its entire postulated time/temperature history, is less severe than the current composite profile.

For containment pressure conditions, the SGR/Uprate accident transient peak pressure is increased by approximately 1%, from 55.91 psia to 56.5 psia; SGR/Uprate average pressures are always less than the current average pressures. A sampling of electrical equipment EQDPs indicates a test peak pressure substantially higher than the SGR/Uprate peak pressure (i.e., over 50% margin). When considering the SGR/Uprate peak pressure there is a slight decrease in margin; however, it is expected that all EQDPs would remain higher than the IEEE Standard 323-1974 required margin. Therefore, the slight increase in accident containment peak pressure is insignificant.

For the MSLT, both the peak pressure and temperature after an accident increase following the SGR/Uprate. The SGR/Uprate peak pressure increases about 1.8 psia. The time duration at or near the peak pressure is less in the current postulated condition. In addition, equipment is generally qualified for a much higher level of inside containment peak pressure. Therefore, there should be no adverse impact on equipment qualification.

The SGR/Uprate MSLT peak temperature increases by 4°F. However, the time duration at or near the peak temperature is also less. Because of this much shorter duration at peak temperature and considering thermal lag, the actual surface temperature for the qualified equipment would be less (i.e., comparable to the previously demonstrated qualification). Therefore, there should be no adverse impact on equipment qualification.

2.23.2.3 Mechanical Equipment Process Temperature Evaluation

For post SGR/Uprate condition, an initial sample review of the process conditions did not reveal any process temperature change greater than +10% from the actual temperature. Because of the large difference between the pre-SGR/Uprate process temperature and the actual temperature withstand capability of the non-metallic materials, the increase in the post-SGR/Uprate process temperature will have no adverse impact on mechanical equipment qualification.

2.23.3 Acceptance Criteria

Safety related equipment required to perform safety functions shall be demonstrated to be capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement, which is embodied in General Design Criteria 1, 2, 4 and 23 of Appendix A and Section III and XI of Appendix B to 10CFR50, is applicable to all safety related equipment and Regulatory Guide 1.97 instruments located inside and outside containment. More detailed requirements and guidance relating to the methods and procedures for demonstrating this capability have been set forth in 10CFR50.49.

The design, operation, and functional qualifications of equipment important to safety described in the FSAR, and as affected by the SGR/Uprate, were evaluated against the acceptance criteria. The results of these evaluations are described in Section 2.23.4.

2.23.4 Results

The SGR/Uprate results in small changes to the qualification parameters and in some instances a small reduction in qualification margin. Safety-related structures, systems and components remain qualified for the intended service under SGR/Uprate conditions.

The SGR/Uprate alters equipment environmental conditions. Safety related equipment was evaluated with regard to changes in the established EQ profiles including increased exposure to TID. Accident and post accident temperature and pressure profiles were evaluated for equipment inside containment and the main steam line tunnel. In addition, the SGR/Uprate does not adversely affect the existing environmental qualification of safety-related equipment with regard to chemical exposure, spray, and submergence.

SGR/Uprate does not adversely affect the equipment qualification. The design limits of safety-related equipment are conservative and bound the new environmental conditions imposed by the SGR/Uprate. All safety-related components are designed to perform required functions with consideration given to the normal and post-accident service environments. Although a slightly higher onsite radiation dose will result from SGR/Uprate and the additional consideration of daughter effects, this will not unacceptably degrade normal or post accident service environments. Any increased radiation levels due to SGR/Uprate are small and fully bounded by the existing design capabilities.

The changes in accident condition pressures in the containment and in the main steam line tunnel are insignificant and will not result in an appreciable increase to the driving force for moisture intrusion into cable insulation systems/electrical terminations or mechanical assemblies.

2.23.5 Conclusions

The effect of the SGR/Uprate on equipment qualification as described in the FSAR 3.11 indicates that there are no changes in environmental conditions, except for exposure to TID, and the accident/post-accident temperature /pressure profile inside the containment and main steam line tunnel. These changes will not adversely impact the equipment qualification or design bases of equipment necessary for safe plant operations. They are bounded by equipment design limits and will not adversely diminish the capability of safety-related equipment in performing their intended safety function. The radiological TID was either enveloped by the results of previous design bases radiological analysis or was within the threshold limit for which the individual component or equipment was qualified. These results are consistent with the current design and licensing bases.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation of the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.23.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications

3/4.3.3	Monitoring Instrumentation
Table 3.3-6	Radiation Monitoring Instrumentation for Plant Operations
3/4.4.8	Specific Activity
5.2.2	Containment Design Pressure and Temperature
6.11	Radiation Protection Program

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2)" dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

8. NUREG-0588, "Interim Staff Position on Electrical Equipment," dated December 1979

9. 10CFR20, "Standards For Protection Against Radiation"

10. 10CFR50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"

11. RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." (Rev. 2, 6/74).

12. RG 1.49, "Power Levels of Nuclear Power Plants," (Rev. 1, 12/73)

13. RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," (Rev. 0, 11/74)

14. RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environmental Conditions Following an Accident," (Rev. 2)

15. IEEE Standard 323-1974, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations

TABLE 2.23-1

Containment Accident Environmental Parameters

Environmental Parameter	Current Conditions	SGR/Uprate Conditions	Evaluation Comment
Accident Duration	365 days	365 days	No change
Humidity	100%	100%	No change
Submergence	228.3 feet	228.5 feet	No significant change
Boron Concentration	2400 – 2600 ppm	2400 – 2600 ppm	No change
pH	8.5 – 11	8.5 – 11	No change
Composite (LOCA/MSLB) Temperature Profile	FSAR Figure 3.11.4-1	LR Figure 2.23-1 (herein)	LR Figure 2.23-2 super-imposes pre-SGR/Uprate profile over post-SGR/ Uprate profile.
Peak Temperature	368°F	364.4°F	Post-SGR/Uprate is less than pre-SGR/Uprate
Average Temperature 0 – 24 hrs: 24 Hrs – 365 days: 0 – 365 days:	 189.6°F 133.23°F 133.38°F	 188.0°F 130.5°F 130.66°F	All post-SGR/Uprate aver-age temperatures are less than the pre-SGR/Uprate average temperatures.
Composite (LOCA/MSLB) Pressure Profile	FSAR Figure 3.11.6-1	LR Figure 2.23-2 (herein)	Pre- and post-SGR/Uprate profiles are comparable.
Peak Pressure	55.91 psia	56.50 psia	Post-SGR/Uprate is only 0.59 psi (or 1.06%) more then pre-SGR/Uprate, which is insignificant.
Average Pressure: 0 – 24 hrs: 24 hrs – 365 days: 0 – 365 days:	 33.02 psia 18.56 psia 18.60 psia	 30.38 psia 18.36 psia 18.40 psia	All post-SGR/Uprate aver-age pressures are less than pre-SGR/Uprate average pressures.

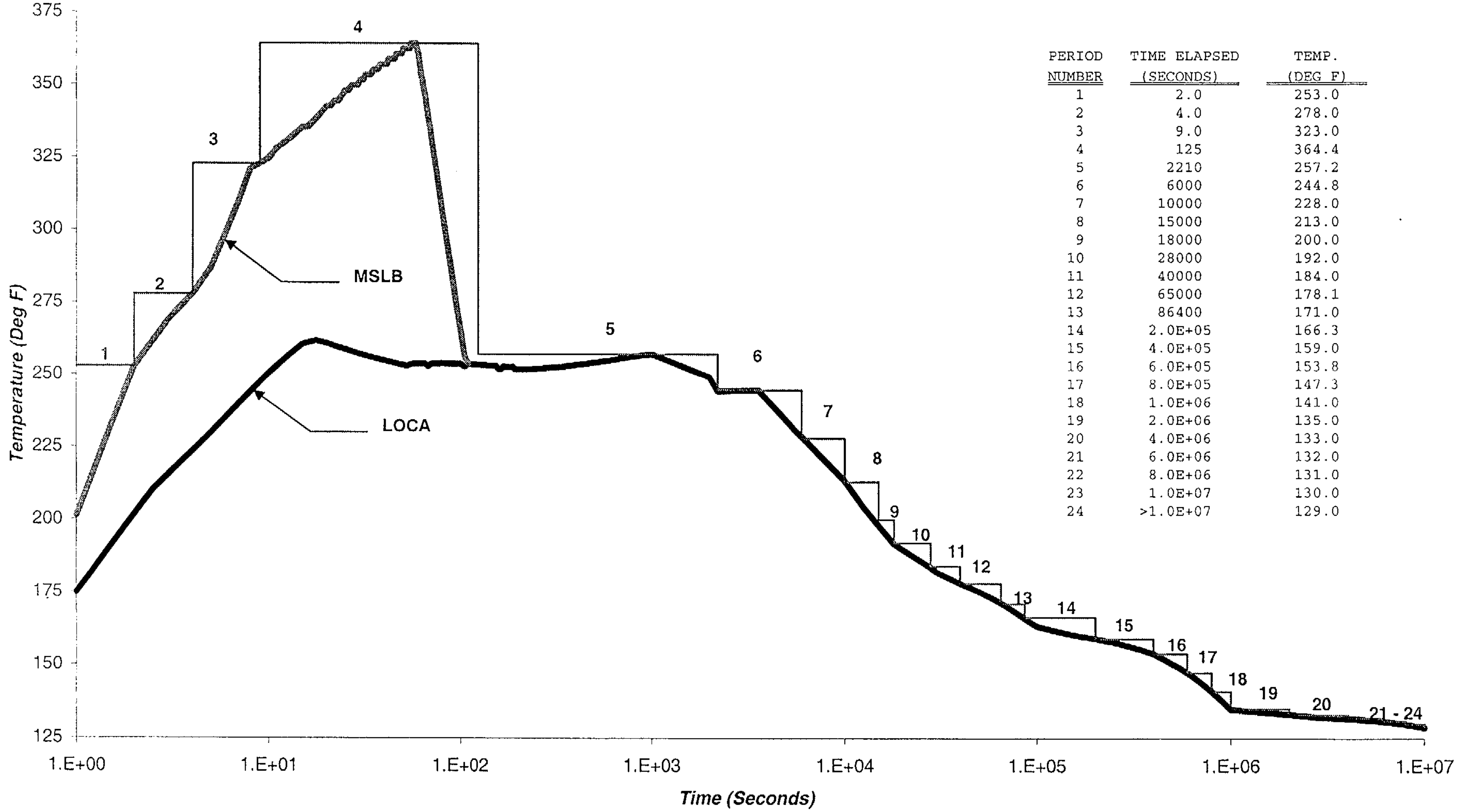
TABLE 2.23-2

Main Steam Line Tunnel Accident Environmental Parameters

Environmental Parameter	Pre-SGR/Uprate	Post-SGR/Uprate	Evaluation Comment
Peak Temperature	437°F from ~ 130 seconds to 640 seconds	441°F at 72 seconds	Post-SGR/Uprate is only 4°F (or 0.92%) more then pre-SGR/Uprate, but peak is present for a shorter time period.
Temperature Profile	FSAR Figure 3.11.4-4	LR Figure 2.23-3	See above comment
Peak Pressure	18 psia at 0.75 seconds	19.8 psia at 0.4 seconds	Post-SGR/Uprate is only 1.8 psi (or 10%) more then pre-SGR/Uprate, but peak is present for a shorter time period.
Pressure Profile	FSAR Figure 3.11.6-3	LR Figure 2.23-4	See above comments
Submergence	265.46 feet	265.32 feet	Post-SGR/Uprate flood level is slightly less than pre-SGR/Uprate flood level.

FIGURE 2.23-1

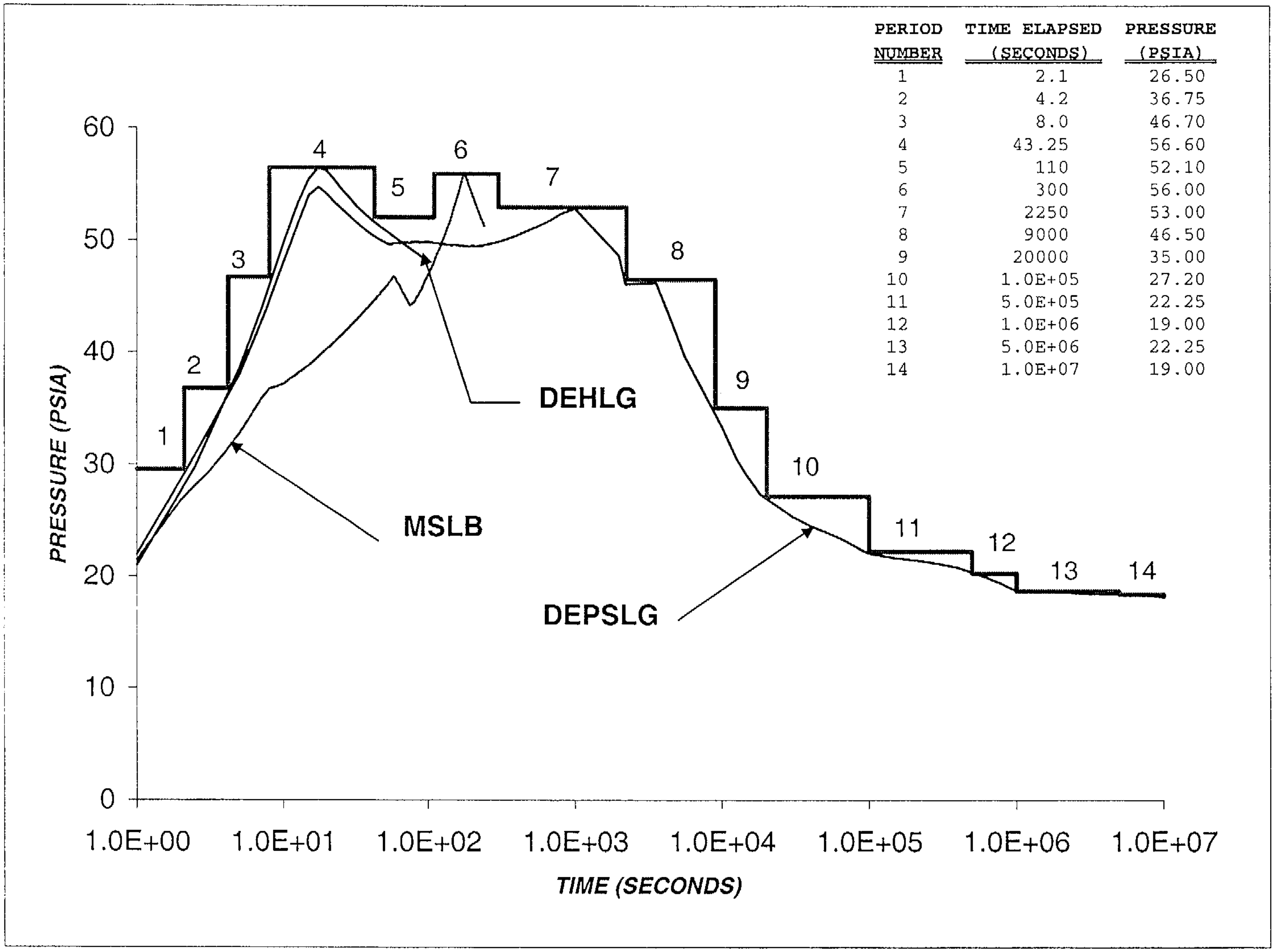
DBA Temperature Profile Inside Containment (Combined LOCA/MSLB)
for Equipment Qualification



PERIOD NUMBER	TIME ELAPSED (SECONDS)	TEMP. (DEG F)
1	2.0	253.0
2	4.0	278.0
3	9.0	323.0
4	125	364.4
5	2210	257.2
6	6000	244.8
7	10000	228.0
8	15000	213.0
9	18000	200.0
10	28000	192.0
11	40000	184.0
12	65000	178.1
13	86400	171.0
14	2.0E+05	166.3
15	4.0E+05	159.0
16	6.0E+05	153.8
17	8.0E+05	147.3
18	1.0E+06	141.0
19	2.0E+06	135.0
20	4.0E+06	133.0
21	6.0E+06	132.0
22	8.0E+06	131.0
23	1.0E+07	130.0
24	>1.0E+07	129.0

FIGURE 2.23-2

DBA Pressure Inside Containment (Combined LOCA/MSLB)
for Equipment Qualification



PERIOD NUMBER	TIME ELAPSED (SECONDS)	PRESSURE (PSIA)
1	2.1	26.50
2	4.2	36.75
3	8.0	46.70
4	43.25	56.60
5	110	52.10
6	300	56.00
7	2250	53.00
8	9000	46.50
9	20000	35.00
10	1.0E+05	27.20
11	5.0E+05	22.25
12	1.0E+06	19.00
13	5.0E+06	22.25
14	1.0E+07	19.00

FIGURE 2.23-3

Temperature Profile for Main Steam Tunnel
(1.4 ft^2 MSLB at 102% Power)

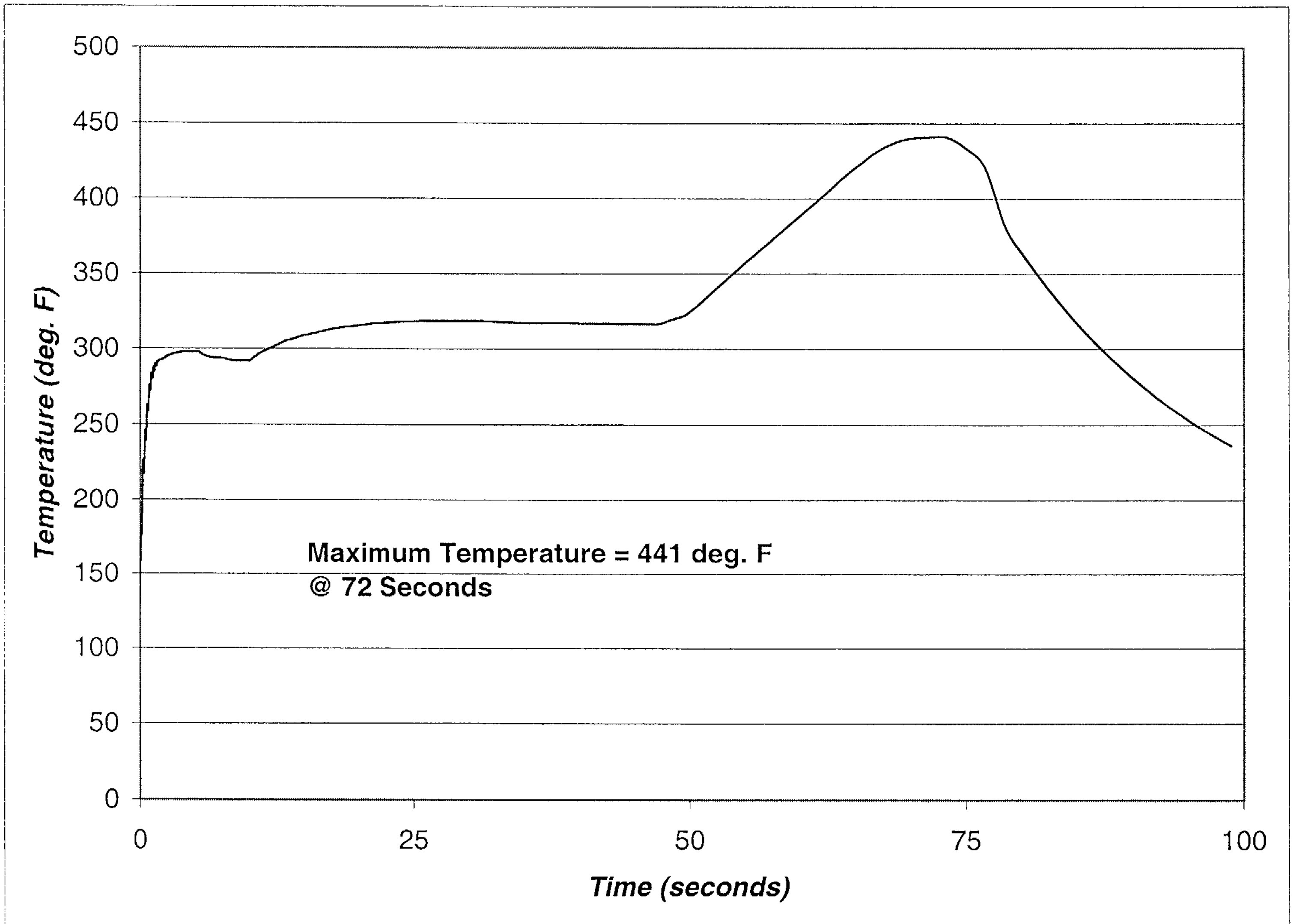


FIGURE 2.23-4

Pressure Profile for Main Steam Tunnel
(1.4 ft^2 MSLB at 102% Power)

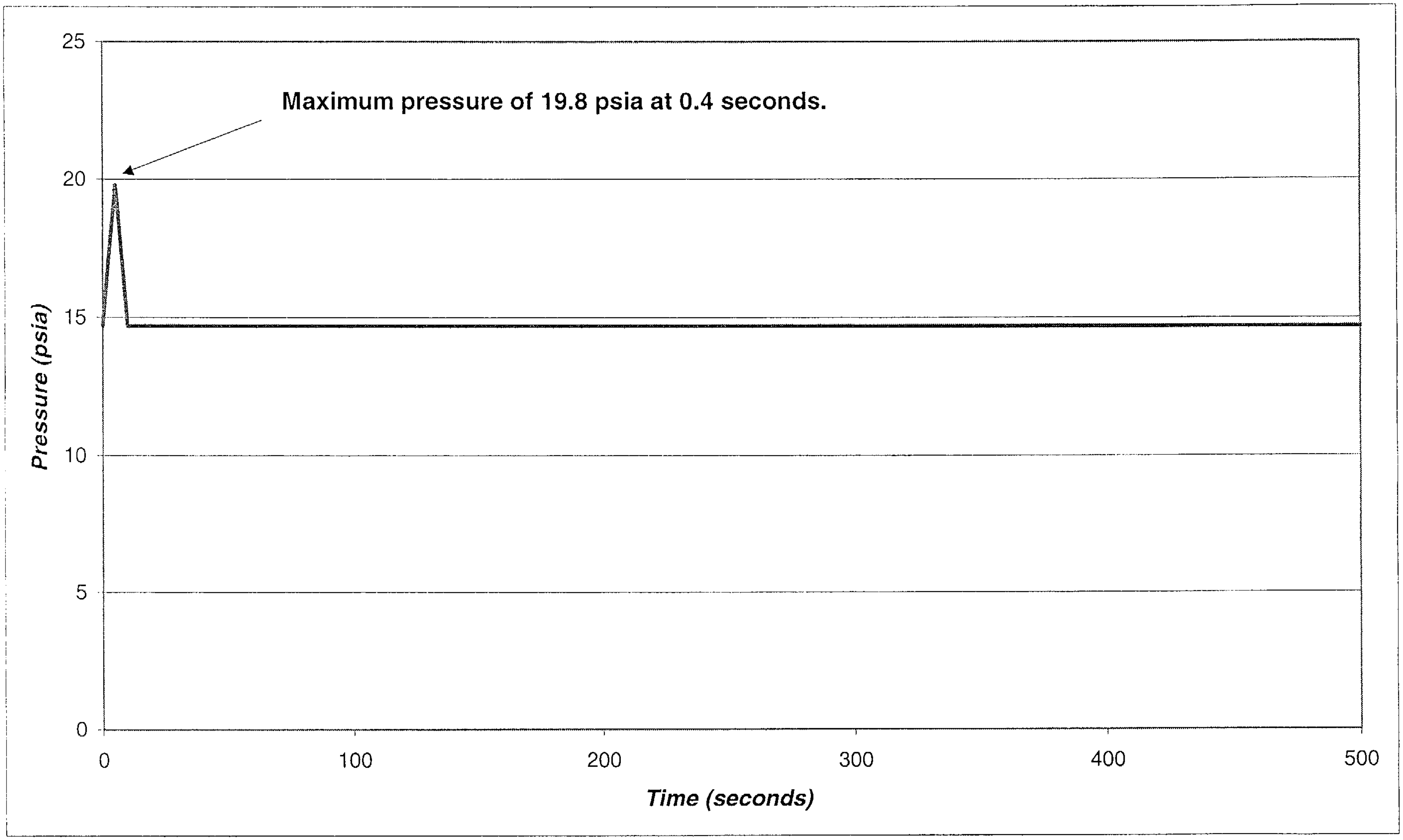
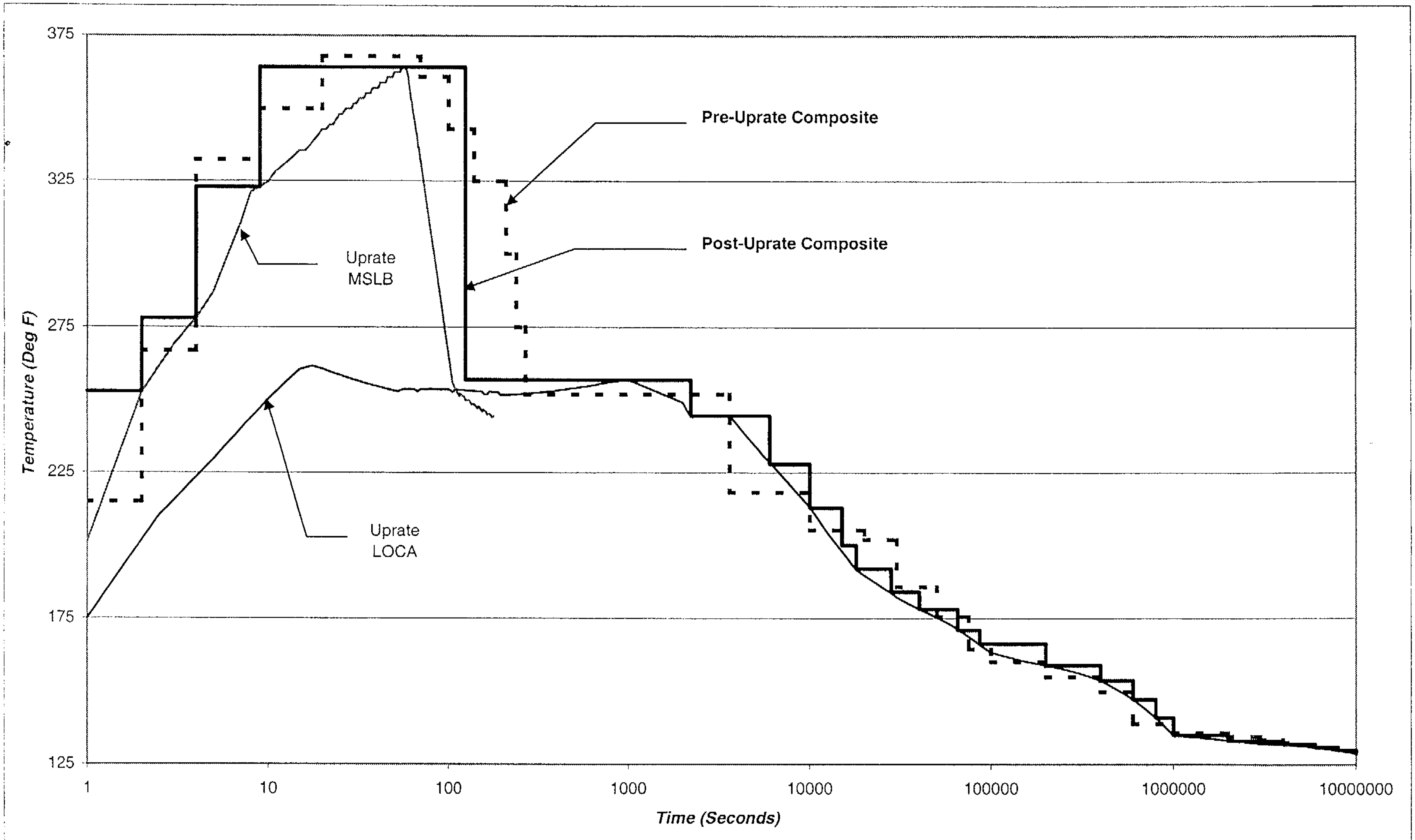


FIGURE 2.23-5

Pre-Uprate and Post-Uprate DBA Composite Temperature Comparison



2.24 Containment and Subcompartment Analyses

Containment and Subcompartment Analyses have been evaluated to demonstrate the structural integrity of containment, its internal structures and equipment supports, and the steam tunnel when subjected to dynamic localized pressurization with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.24.1 Introduction and Background

As described in the Final Safety Analysis Report (FSAR), the Containment Loss-of-Coolant-Accident (LOCA) and Main Steam Line Break (MSLB) analyses are performed to demonstrate that the containment structure can withstand the consequences of the most severe LOCA or MSLB and remain capable of performing required functions in accordance with the FSAR. In addition, the analyses generate pressure and temperature curves used in the qualification of the safety-related equipment inside containment. Equipment qualification is addressed in LR Section 2.23 "Equipment Qualification". The maximum containment pressure also establishes the test pressure at which containment leakage is monitored, as described in Technical Specification 6.8.4.K.

MSLB analysis inside/outside containment is performed to also demonstrate that the steam tunnel can withstand the consequences of the most severe MSLB and that the steam tunnel remains capable of performing required functions in accordance with the FSAR. This analysis generates the pressure and temperature curves used in the qualification of the equipment inside the steam tunnel.

The subcompartment analyses are performed to demonstrate the structural integrity of containment internal structures when subjected to dynamic localized pressurization effects that occur during the first few seconds following a design basis pipe break accident. After the postulated rupture, the affected subcompartment pressure builds up at a faster rate than the overall containment pressure, thus imposing differential pressures across the internal walls of the structure, as well as forces and moments on the equipment supports.

2.24.2 Description of Analyses and Evaluations

2.24.2.1 <u>Containment Analysis</u>

Changed process conditions due to SGR/Uprate affect the containment analyses by increasing the mass and energy blowdown from pipe breaks. The re-analysis utilized the following conservative assumptions:

- The nitrogen contained in all three accumulators is released to the containment within one second following the break (was not considered in the previous LOCA analyses).
- A spray actuation time of 58.4 seconds was used, versus 57.27 seconds in the previous LOCA analyses and 41.59 seconds in the MSLB analyses.
- A minimum containment spray flow rate of 1730 gpm was used, versus 1832 gpm in the previous LOCA and MSLB analyses.

- The fan cooler heat removal rates used correspond to 1300 gpm service water flow, versus 1360 gpm in the previous LOCA and MSLB analyses; 1 cooling coil per safety train is assumed to be plugged; and a fouling factor of 0.001 is used. In addition, the heat generated by the fan motor is considered in the analysis.

The new analyses determined two LOCA cases to be limiting breaks: 1) a double-ended pump suction leg guillotine break (DEPSLG) with minimum safety injection and 2) a double-ended hot leg guillotine break (DEHLG). The hot leg break was analyzed to determine the maximum pressure and temperature inside containment during the initial short-term blowdown phase. The pump suction break yields the highest energy flow rates during the post-blowdown period; therefore it is used to establish the long-term pressures and temperatures. For completeness, the case of a double-ended pump suction leg guillotine break (DEPSLG) with maximum safety injection was also evaluated.

The containment was also analyzed for the postulated MSLB. To ensure that the MSLBs having the worst consequences were identified, the spectrum of break types identified below was considered.

- Double-ended ruptures (1.4 ft^2) at 102%, 70%, 30% and 0% power
- Split rupture (0.687 ft^2) at 102% power
- Split rupture (0.675 ft^2) at 70% power
- Split rupture (0.666 ft^2) at 30% power
- Split rupture (0.588 ft^2) at 0% power

A sensitivity study for the effect of initial pressure and humidity show that to maximize the calculated peak containment pressure, the maximum initial containment pressure of 16.3 psia and the minimum humidity of 20% should be used. For maximum temperature, the minimum containment pressure of 14.66 psia and the maximum humidity of 75% are conservative.

The following (18) specific MSLB cases were considered:

Description
0% power, cooling train failure, maximum pressure
0% power, cooling train failure, maximum temperature
0% power, MFIV failure, maximum pressure
0% power, MFIV failure, maximum temperature
30% power, cooling train failure, maximum pressure
30% power, cooling train failure, maximum temperature
30% power, MFIV failure, maximum pressure
30% power, MFIV failure, maximum temperature
70% power, cooling train failure, maximum pressure
70% power, cooling train failure, maximum temperature
70% power, MFIV failure, maximum pressure
70% power, MIFV failure, maximum temperature
102% power, cooling train failure, maximum pressure
102% power, cooling train failure, maximum temperature
102% power, MFIV failure, maximum pressure

102% power, MFIV failure, maximum temperature
30% power, split break case, cooling train failure, maximum pressure
30% power, split break case, cooling train failure, maximum temperature

FSAR Table 6.2.1-4 shows that a full double-ended break at a given power is more severe than a corresponding split break. Consequently, the double-ended breaks various levels were analyzed for the SGR/Uprate. Two split break cases were analyzed to demonstrate that the split break cases are not the controlling cases.

Mass and energy release data used in the analysis of double-ended breaks at each of the four postulated power levels reflect the failure of the faulted loop's main steam isolation valve (MSIV). In addition to this failure assumption, the SGR/Uprate analyses considered the following single active failures (SAFs):

- Failure of a main feedwater isolation valve (MFIV); or
- Failure of a feedwater flow control valve (MFCV); or
- Failure that leaves only one cooling train available for heat removal.

The dryout times for the MFCV cases are significantly lower than dryout times for the MFIV. For this reason, the MFCV failure cases are enveloped by the MFIV failure cases and the MFCV cases are not analyzed.

The CONTEMPT-LT/26 computer code was used for the LOCA analyses. The CONTEMPT-LT/28 computer code was used for the MSLB analyses.

2.24.2.2 Subcompartment Analysis

The Reactor Building subcompartments (steam generator compartments, pressurizer compartment and reactor cavity) were evaluated for the appropriate design basis break to determine the peak pressure differential. The following design basis pipe breaks were evaluated to determine the impact of SGR/Uprate on the results from previous analyses:

1. Reactor Cavity: 150 in^2 Cold Leg Break, 150 in^2 Hot Leg Break.

2. Steam Generator Compartments: Double-Ended Cold Leg Guillotine Break, Double-Ended Pump Suction Leg Guillotine Break, and Double-Ended Hot Leg Guillotine Break.

3. Pressurizer Compartment: Double-Ended Pressurizer Surge Line Break and Pressurizer Spray Line Break. Since the pressurizer compartment is immediately adjacent to the Loop 2 Steam Generator compartment and the opening between the compartments is fairly large, the maximum pressure inside the Loop 2 SG compartment due to Double-Ended Pump Suction Leg Guillotine break was found to be limiting for the pressurizer compartment.

2.24.2.3 Steam Tunnel

The following MSLBs inside the Steam Tunnel have been evaluated with consideration given to the SGR/Uprate at 101 and 70 percent power:

- A 1.4 ft^2 break which represents the largest break size
- A 0.1 ft^2 break which represents the smallest break size
- A 0.7 ft^2 break which represents an intermediate case.

The results of the postulated MSLBs in the steam tunnel are significantly higher than those for the feedwater line breaks. Therefore only steam line breaks have been analyzed. The COMPRESS computer code was used in the SGR/Uprate for the MSLB analyses outside containment.

2.24.3 Acceptance Criteria

The acceptance criteria reviewed in this section are as follows:

- The maximum containment pressure and temperature shall remain less than the pressure and temperature limits in Technical Specification 5.2.2. The containment pressure at 24 hours after the LOCA shall be less than half the calculated peak pressure.

- The maximum pressure in a containment subcompartment shall remain less than the design pressure subsequent to a LOCA.

- The maximum pressure in the steam tunnel shall remain less than the design pressure subsequent to a Main Steam Line Break (MSLB).

2.24.4 Results

2.24.4.1 Containment Analysis

Based upon the LOCA analyses, for the long term case, the peak pressure and temperature are calculated to be 54.8 psia (40.1 psig) and 261.8°F for the DEPSLG-LOCA. For the short term case, the DEHLG-LOCA, the peak calculated pressure and temperature are 56.5 psia (41.8 psig) and 270.2°F.

As for the long term containment pressure transient at 24 hours, the containment pressure is calculated to be 22.5 psia (7.8 psig), which is less than half the peak calculated pressure.

Tables 2.24.4-1 and 2.24.4-2 present the results of the eight MSLB cases analyzed. The most limiting MSLB cases are the full double-ended break at 30% power with MFIV failure for maximum pressure (41.3 psig) and the full double-ended break at 102% power with either one cooling train or MFIV failure for the maximum temperature (364.4°F). The MSLB cases analyzed also include the SAF of the MSIV. SGR/Uprate peak

LOCA/MSLB response conditions (used for equipment qualification) are presented in BOP LR Section 2.23.

The results of the LOCA and MSLB re-analyses are less than the pressure and temperature limits as described in Technical Specification Section 5.2.2.

Table 2.24.4-1

Containment Pressure/Temperature
Full D/E MSLB - Cooling Train Failure

Power Level (%)	102	70	30	0
Peak Pressure (psig)	35.1	36.1	38.3	37.9
Peak Temperature (°F)	364.4	361.3	359.3	355.7
Time of Peak Pressure (sec)	108.2	124.7	149.2	291.5

Table 2.24.4-2

Containment Pressure/Temperature
Full D/E MSLB - Main Feedwater Isolation Valve Failure

Power Level (%)	102	70	30	0
Peak Pressure (psig)	37.8	39.2	41.3	39.8
Peak Temperature (°F)	364.4	361.3	359.3	355.7
Time of Peak Pressure (sec)	132.7	150.2	176.2	365.0

2.24.4.2 Subcompartment Analysis

Leak Before Break (LBB) Methodology has been applied to the pipe rupture evaluations for the SGR/Uprate. The application of LBB methodology, which is reflected in the current licensing bases, eliminated the dynamic effects of postulated primary Reactor Coolant Loop pipe ruptures from the design basis for the Reactor Building. However, LBB does not eliminate breaks in the primary side auxiliary piping. As a result, the dynamic effects of the 150 in^2 cold/hot leg break, DECL break, DEPSLG break and DEHLG break that were considered in the original design basis for the Reactor Coolant System (RCS) or Reactor Building, do not have to be considered for the re-analysis of SGR/Uprate effects. The analyses for SGR/Uprate must consider the dynamic effects of postulated breaks in only the surge line, RHR line, spray lines, feedwater lines, steam lines and the accumulator nozzles.

Reactor Cavity

The break sizes associated with the postulated surge line, the RHR lines, and accumulator nozzle breaks are less than the 150 in^2 design basis pipe breaks. The lower mass and energy releases from these smaller RCS breaks offset any process condition changes (increased temperatures, pressures, flows) associated with SGR/Uprate. The surge line, RHR lines and accumulator lines are outside the reactor vessel cavity region and postulated breaks will result in minimal asymmetrical pressurization in the reactor cavity region. Consequently, the mass and energy release for the design basis 150 in^2 breaks bound those for SGR/Uprate break effects.

Steam Generator Compartment

The break sizes for the surge line, RHR lines and accumulator nozzles under the SGR/Uprate are significantly smaller than the double-ended rupture in hot-leg, cold-leg and pump suction previously utilized in SG compartment analyses. These smaller breaks result in lower mass and energy releases that offset any configuration changes within the SG compartments. Therefore, the pressure, forces and moments used in SG compartment design remain bounding for the SGR/Uprate.

Pressurizer Compartment

Surge line and spray line breaks are postulated in the pressurizer compartment to evaluate subcompartment pressurization effects under SGR/Uprate. Since LBB methodology is only applied to large RCS piping, postulated breaks in these smaller lines are considered within this compartment.

The release rates from breaks in the spray line under the SGR/Uprate have increased. These release rates are, however, bounded by the values from the postulated surge line breaks.

The forces and moment on the pressurizer due to surge line break are much lower than those due to pump suction break because of the location of the break. The surge line break is located at the bottom of the pressurizer giving a much smaller moment arm. Therefore, the forces and moments on the pressurizer are bounded by the pump suction leg break in the SG compartment.

2.24.4.3 Steam Tunnel

The results of the MSLB analyses in the steam tunnel showed that the maximum temperature is calculated to be 441°F due to a 1.4 ft^2 break at 102% power and the maximum pressure is calculated to be 5.1 psig due to a 1.4 ft^2 break at 70% power.

2.24.5 Conclusions

2.24.5.1 Containment Analysis

The maximum containment pressure during a LOCA will increase slightly as a result of SGR/Uprate. The maximum containment pressure, following SGR/Uprate, is calculated to be 41.8 psig which is less than the containment design pressure of 45 psig and is therefore acceptable.

The analysis for a MSLB inside containment indicates that the maximum containment pressure is 41.3 psig for a very short duration and the containment peak temperature is 364.4°F. These calculated values are less than the containment design pressure of 45 psig and containment design temperature of 380°F and are therefore acceptable.

2.24.5.2 Subcompartment Analysis

The SGR/Uprate Project does not impact any of the current sub-compartment analyses that established the current peak pressure differentials for various compartments.

2.24.5.3 Steam Tunnel

The maximum pressure in the steam tunnel following SGR/Uprate is calculated to be 5.1 psig. This is less than the steam tunnel design pressure of 16 psig and is therefore acceptable. The SGR/Uprate peak temperature and pressure conditions (used for equipment qualification) are presented in BOP LR Section 2.23.

2.24.7 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.6.1 Primary Containment
   B3/4.6.1.4 Internal Pressure
   B3/4.6.1.6 Containment Structural Integrity
   5.2.2 Design pressure and Temperature
   6.8.4.K Containment Leakage Rate Testing Program

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"

9. RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," (Rev. 2, 6/74)

10. 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants"
Criterion 4, "Environmental and Dynamic Effects Design Bases"
Criterion 13, "Instrumentation and Control"
Criterion 16, "Containment Design"
Criterion 38, "Containment Heat Removal"
Criterion 50 "Containment Design Basis"

2.25 Post-LOCA Hydrogen Control System

The Harris Nuclear Plant (HNP) Post-LOCA Hydrogen Control System has been evaluated to determine its performance capabilities for plant operation with the Model Delta 75 replacement steam generators (RSGs) at the uprated NSSS power level of 2912.4 MWt.

2.25.1 Introduction and Background

After a loss-of-coolant accident (LOCA), hydrogen may accumulate within the containment as a result of (1) metal-water reaction between the zirconium fuel cladding and the reactor coolant, (2) radiolytic decomposition of the water in the core and containment sump, (3) corrosion of metals by emergency core cooling and containment spray solutions, and (4) the release of hydrogen dissolved in the reactor coolant. As described in the Final Safety Analysis Report (FSAR), Section 6.2.5, the following components are used to monitor and control potential buildup of hydrogen within the containment.

The post-accident hydrogen monitoring system consists of containment sampling valve manifolds, containment isolation valves, hydrogen analyzers, a remote control panel, a sample dilution panel, and a sample return line. The system is designated Class 1E and designed to seismic Category 1 requirements. The system provides continuous indication and recording of the containment hydrogen concentration.

As described in FSAR Section 6.2.5.1.1, primary control of containment hydrogen is provided by two redundant, independently powered electric hydrogen recombiners located inside the containment. These units are designed to seismic Category I and safety Class 2 standards, each powered by a separate Class 1E safeguard bus. The containment atmosphere is circulated through the recombiners by natural convection where the hydrogen bearing gases are heated to a temperature sufficient to cause recombination. The recombiners consist of an inlet preheater section, a heater-recombiner section, and a discharge mixing chamber that lowers the exit temperature of the air.

A backup method of controlling containment hydrogen is provided by the post-accident hydrogen purge system. The post-accident hydrogen purge system consists of a purge make-up penetration line, an exhaust penetration line, and a filtered exhaust system. The filtered exhaust system includes a demister, an electrical heating coil, a medium efficiency filter, a HEPA pre-filter, charcoal adsorber, HEPA after-filter, and a centrifugal fan that discharges to the stack. The only portion that meets safety Class 2 and seismic Category 1 requirements are the lines penetrating the containment up to, and including the first containment isolation valve outside containment.

Mass diffusion of the hydrogen from the generation source(s) within the Containment Building provides mixing with the rest of the containment atmosphere. This natural circulation when coupled with the active mixing, as provided by the Containment Fan

Coolers and Containment Sprays, assures proper uniform mixing of the hydrogen to preclude local concentrations exceeding four volume percent.

The Steam Generator Replacement and Power Uprate Project (SGR/Uprate) does not change Post-LOCA Hydrogen Control System design functions. The Post-LOCA Hydrogen Control System remains capable of satisfying regulatory commitments in accordance with the existing FSAR following the SGR/Uprate.

Configuration Changes:

There are no configuration changes to the Post-LOCA Hydrogen Control System required to support the SGR/Uprate.

Revised Process Conditions:

The SGR/Uprate will alter the post-accident environmental conditions inside the Containment, increasing the initial inventory of hydrogen and also resulting in an increased generation/accumulation rate.

The impact of the increased hydrogen generation/accumulation to the Post-LOCA Hydrogen Control System is addressed in Section 2.25.2.

2.25.2 Description of Analyses and Evaluations

The Post-LOCA Hydrogen Control System was evaluated to ensure that, following the SGR/Uprate, the system remains capable of performing its required functions in accordance with existing licensing bases as specified in the FSAR.

The hydrogen generation study performed as part of the design basis review, shows that the containment hydrogen percentage has increased as a function of time. The revised analyses show approximately 9.5×10^4 scf of hydrogen production at 12 days after a LOCA as opposed to 8.0×10^4 scf. Relative to the results in the current FSAR, this increase in the hydrogen accumulation rate is attributed to the following:

1. Independent of SGR/Uprate, CP&L is in the process of incorporating a containment temperature increase into the H_2 calculation.

2. Independent of SGR/Uprate, a larger inventory of zirconium (the current FSAR evaluation is based on 36,795 lb. of zirconium and the new analysis is based on 39,600 lb. of zirconium),

3. Independent of SGR/Uprate, a larger conservative percentage of reaction (5 x 1% in accordance with the requirements of 10 CFR 50.46(b)(3) vs. the 5 x 0.3% that Westinghouse generically showed in the WCAP to be applicable for their fuel), and

4. With the implementation of SGR/Uprate, the methodology for determining the hydrogen generated from radiolysis has been updated. This analysis utilizes the new methodology provided in NUREG-0800. Using this new method, the hydrogen produced by radiolysis at 12 days has increased (i.e., 3.4 x 10^4 sfc vs. 3.0 x 10^4 sfc).

FSAR Figure 6.2.5-6 "Hydrogen Accumulation vs. Time After Start of LOCA" has been revised and is attached to this report.

The post-accident environmental conditions inside the Containment are changed by the SGR/Uprate and the amount of time it takes for containment hydrogen concentrations to reach 3% has been reduced, but still exceeds 4 days. The SGR/Uprate does not adversely affect the ability of the Post-LOCA Hydrogen Control System to monitor and control hydrogen concentration in the containment following a LOCA. The increased hydrogen accumulation rate can be accommodated since actuation of a single hydrogen recombiner when the containment hydrogen concentration reaches 3 volume % is sufficient to maintain the hydrogen concentration less than 4% volume percent. The EOPs contain steps to align the hydrogen monitors and the hydrogen recombiners in a timely manner prior to reaching 3 volume %.

The results of the evaluation of the Post-LOCA Hydrogen Control System, and its individual components, to satisfy applicable design and licensing bases in accordance with the FSAR are presented in Section 2.25.4. Acceptance criteria, relevant to the Post-LOCA Hydrogen Control System, are identified in Section 2.25.3.

2.25.3 Acceptance Criteria

The licensing bases for the Post-LOCA Hydrogen Control System are described in the FSAR. The Post-LOCA Hydrogen Control System is designed to monitor and control the potential buildup of hydrogen within containment following a LOCA. The system must maintain the concentration of hydrogen in the containment below four percent of volume.

The design, operation, and functional capabilities of the Post-LOCA Hydrogen Control System described in the FSAR, and as affected by the SGR/Uprate, were evaluated against the acceptance criteria. The results of these evaluations are described in Section 2.25.4.

2.25.4 Results

The SGR/Uprate does not change the Post-LOCA Hydrogen Control System, and its intended design functions, as described in the FSAR. There are no configuration changes required for the Post-LOCA Hydrogen Control System under SGR/Uprate conditions and any changes to process conditions (e.g., an increased hydrogen accumulation rate) are within the capabilities of the existing design.

The Technical Specifications were evaluated with consideration given to the SGR/Uprate. The Technical Specifications, which are related to the Post-LOCA Hydrogen Control System are not impacted by the SGR/Uprate. The Hydrogen Control System will continue to satisfy required functions.

The Hydrogen Control System components were evaluated to determine their compatibility with the SGR/Uprate. System components including the electric hydrogen recombiners, exhaust system, valves, piping, and hydrogen analyzers are adequate for the SGR/Uprate conditions. The SGR/Uprate does not adversely affect Post-LOCA hydrogen mixing. The components are qualified for operation in their respective post-accident service environments inside the Containment and RAB. The additional Post-LOCA hydrogen accumulation following the SGR/Uprate is acceptable under present component performance.

2.25.5 Conclusions

Although the SGR/Uprate, decreases the time in which a recombiner must be put into operation, the Post-LOCA Hydrogen Control System design is adequate for the system to continue to limit containment Post-LOCA Hydrogen to less than four volume percent.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation of the Delta 75 RSGs at the current NSSS power level of 2787.4 MWt.

2.25.6 References

1. HNP Final Safety Analysis Report

2. HNP Technical Specifications
   3/4.6.3 Containment Isolation Valves
   3/4.6.4 Combustible Gas Control

3. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983

4. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984

5. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985

6. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986

7. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986

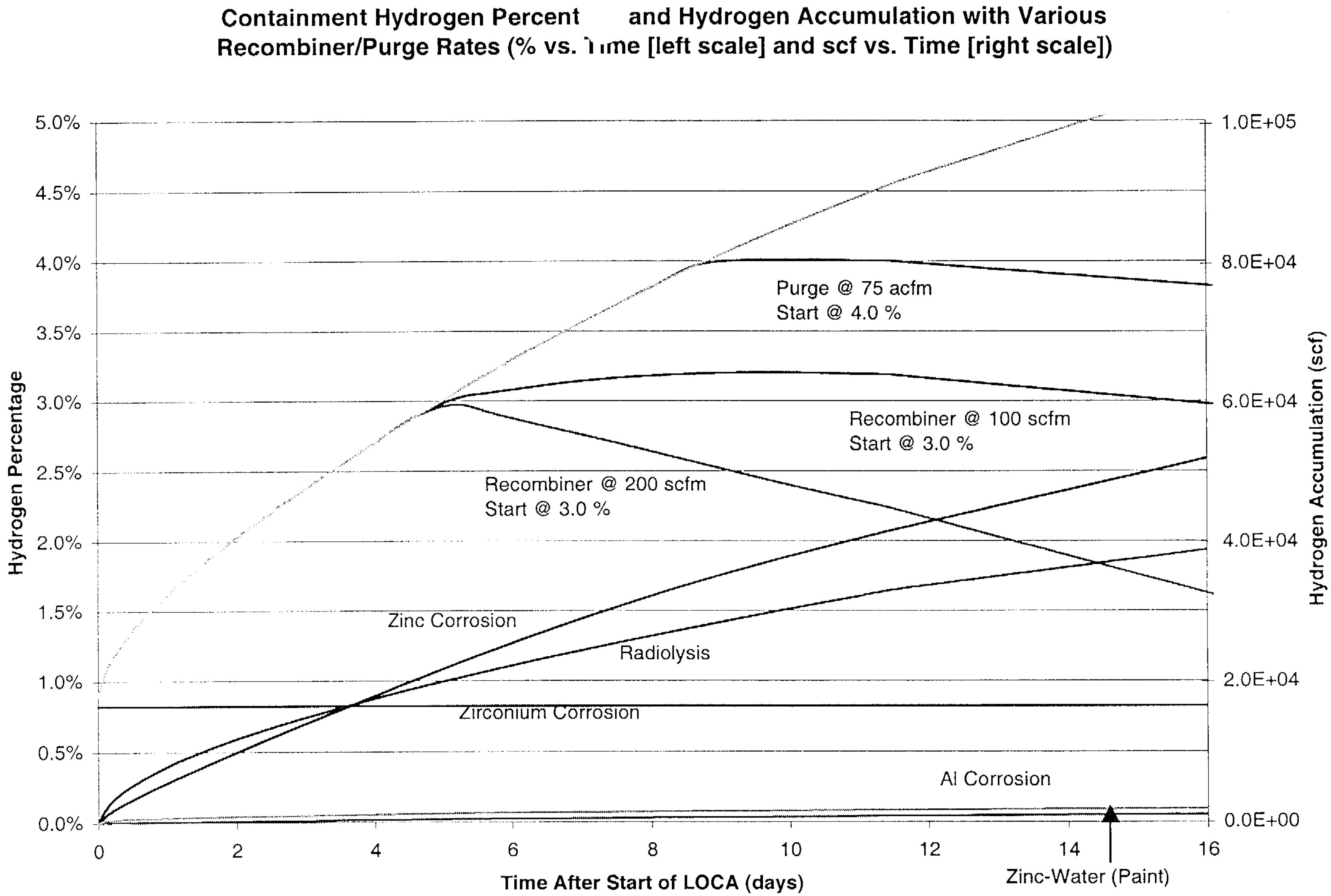


Figure 2.25-1
Proposed
FSAR Figure 6.2.5-6 Hydrogen Accumulation Vs. Time After Start Of LOCA

2.26 Environmental Impact Evaluations

The Harris Nuclear Plant Final Environmental Statement (FES or FES-OL, reference 2.26.4.1) was prepared by the U.S. Nuclear Regulatory Commission (NRC) as an assessment of the environmental impact associated with the operation of the Shearon Harris Nuclear Power Plant, Units 1 and 2. When the FES was issued in October 1983, Unit 2 startup was anticipated to occur several years after the startup of Unit 1. The FES addresses, at several points, operation of Unit 1 alone and then operation of both units. Unit 2 was subsequently cancelled in December 1983. The evaluation that follows will assess changes resulting from steam generator replacement (SGR) and Power Uprate from the conditions evaluated by the NRC as reported in the FES.

The conclusions of the FES are based on NRC review of information provided in the Environmental Report - Operating License Stage (ER-OL, reference 2.26.4.2). The environmental evaluation for Power Uprate provides an assessment of the environmental impact associated with power uprate by comparing the operating parameters established for power uprate with the parameters and conclusions documented in the above referenced reports.

Section 3.1 of the Shearon Harris Nuclear Power Plant Environmental Protection Plan (EPP), Appendix B to the Facility Operating license (NPF-63) (reference 2.26.4.3) states . . . *"licensee may make changes to the station design or operation or perform tests or experiments affecting the environment provided such activities do not involve an unreviewed environmental question and do not involve a change to the EPP."* Section 3.1 requires that an environmental evaluation be prepared and recorded prior to engaging in any activity which may significantly affect the environment. Section 3.1 further states ". . .*A proposed change, test or experiment shall be deemed to constitute an unreviewed environmental question if it concerns: (1) a matter which may result in significant increase in any adverse environmental impact previously evaluated in the FES-OL, environmental impact appraisals, or in any decision of the Atomic Safety and Licensing Board; or (2) a significant change in the effluents or power level; or (3) a matter not previously reviewed and evaluated in the documents specified in (1) of this Subsection, which may have a significant adverse environmental impact."*

2.26.1 Scope of Review

In accordance with the requirements discussed above, an evaluation of the environmental impact of the proposed NSSS power level uprate from 2787.4 MWt to NSSS power level 2912.4 MWt has been performed. The environmental impacts that were addressed in the FES and evaluated for SGR/Uprate include:

- Water Consumption
- Thermal Discharge from the Cooling Tower Blowdown
- Cooling Tower Drift
- Impacts on the Auxiliary Reservoir

- Non-radiological Effluents
- Routine Releases of Radiological Effluents
- Noise Impacts
- Terrestrial Impacts

The following environmental evaluation specifically considers effects on the following parameters:

Circulating Water System

Changes in temperature and rate of Cooling Tower blowdown
Changes in makeup to the Cooling Towers
Changes in amount of Cooling Tower drift

Based on evaluation of the Cooling Tower and service water performance parameters, and review of information contained in the Environmental Report - Operating License Stage and FES relative to impacts associated with the Circulating Water System, the following information is provided.

2.26.2 Summary of Evaluation

The proposed power uprate will result in an increase in Cooling Tower duty of approximately 4.2E+08 BTU/hr over the current operating condition, with a corresponding increase in evaporation, makeup and Cooling Tower blowdown temperature. This heat duty includes a component from the Normal Service Water system, which is not expected to change as a result of SGR/Uprate. However, the increase in Cooling Tower duty from the 6.67E+09 BTU/hr evaluated in the ER-OL (for a single unit) is 2.4E+08 BTU/hr or 3.6 %.

Cooling Tower flowrate does not change as a result of power uprate. However, a concurrent project, retubing the Main Condenser, will result in an increase in the Circulating Water System flow by approximately
4,600 gpm. Cooling Tower drift, which is a small fraction (0.002%) of the total Cooling Tower flowrate (Circulating Water System plus Normal Service Water System) will increase slightly. However, the impact on the production of Cooling Tower drift is negligible.

The average temperature of the Cooling Tower Blowdown is predicted to increase by 0.4°F in the winter and 0.1°F in the summer. These values are based on the average January and July wet bulb temperatures presented in the ER-OL Table 3.4.2-2.

CP&L's original analyses predicted the mixing zone for the Cooling Tower Blowdown to be 120 acres in winter and 20 acres in summer. The FES (Section 5.3.1.2.1) concluded that CP&L's original analysis conducted under extreme temperature conditions was conservative and protective of water quality standards. The analyses were done assuming two units in operation. The FES reported independent analyses that predicted that the mixing zone would remain less than 0.7 acres under all conditions. A comparison of the assumptions used by the NRC and conditions that will exist post uprate are as follows:

Parameter	FES Value	Uprate Value	Change Increase (Decrease)
Normal Water Level	220 ft. MSL	220 ft. MSL	None
Low water level	204.4 ft MSL	215 ft. MSL	10.6 ft.
Elevation at Discharge point	182 ft. MSL	182 ft. MSL	None
Rate of Discharge	42 cfs (two units)	22.4 cfs	(19.6) cfs
Jet diameter	4 ft.	4 ft.	None
Discharge velocity	3.7 fps	1.8 fps	(1.9) fps
Temperature Excess	9°F (July) 32°F (Dec.)	9.7°F (July)[1] 34.8°F (Dec.)[1]	0.7°F (July) 2.8°F (Dec.)

[1] The Increase includes a 2.2°F increase to account for the measured efficiency of Cooling Tower.

Based on the above, the additional heat load to the Harris Lake associated with SGR/Uprate of a single unit does not significantly impact the conclusions of the FES relative to thermal impact. The minimal increase in blowdown temperature associated with SGR/Uprate is also conservative and protective of water quality standards. As discussed in the FES, adequate mixing occurs such that the size of the thermal plume is acceptably small. This conclusion remains valid in view of the fact that the original analyses were done assuming two units in operation.

In addition to the FES, the thermal impact associated with power uprate was evaluated relative to the Shearon Harris Nuclear Power Plant NPDES permit. The North Carolina Department of Environment, Health and Natural Resources issued NPDES Permit No. NC0039586 to Shearon Harris Nuclear Power Plant. The permit was last renewed in July 31, 1996. The NPDES permit specifies a mixing zone of an area no greater than 200 acres. The original NPDES permit contained a requirement to monitor the Cooling Tower blowdown to ensure compliance with the requirements of the mixing zone. However, the monitoring results subsequently led to the deletion of the requirement to monitor blowdown temperature in the NPDES. In view of the conservatism in the original CP&L analyses, the deletion of Unit 2, and the small change in Cooling Tower Blowdown temperature, no difficulty will be encountered in meeting the 200 acre limitation on the size of the mixing zone.

The amount of water required to makeup for forced evaporation from the Cooling Tower is expected to increase. The ER-OL predicted the annual average, forced evaporation at a power level of 100% to be 22.1 cubic feet per second (cfs). The revised comparable value for PUR is 22.8 cfs. The increase in the average forced evaporation loss is 0.7 cfs assuming a 95% capacity factor and annual average meteorology.

The increase (0.7 cfs) is small relative to the total water demand from the operation of Unit 1 and the flow available from the inputs to the Main Reservoir. The total water consumption of 32.2 cfs includes forced evaporation (assuming a capacity factor of 95%), natural evaporation from the reservoirs, seepage, and miscellaneous plant consumption. The total inputs to the Main Reservoir averages 67.6 cfs. Therefore, there is no significant impact on the Main Reservoir.

With regard to downstream water uses, the change is small compared to the total Cape Fear River flow (downstream of the Main Dam) of 3,125 cfs.

The NRC, in FES Section 5.3.2.1 stated:

"... less than 1% of the average flow of the Cape Fear River [3125 cfs] will be used by the plant. Thus the staff's conclusion in the RFES-CP that the consumptive water use by a four-unit plant would not adversely affect other downstream water users is valid for a two-unit plant."

The revised water consumption by the plant is approximately 1.03% of the average Cape Fear River flow.

Based on these points, operation of the plant at SGR/Uprated power conditions is not a significant environmental impact.

2.26.3 Summary of Conclusions

The impacts from SGR/Uprate on Cooling Tower drift, thermal discharges to the Ultimate Heat Sink (Auxiliary Reservoir), non-radiological effluents, noise impacts and terrestrial impacts were determined to be well within the effects evaluated by the FES because the evaluations done in the FES were based on a two unit plant.

Based on the above evaluation, the plant operating parameters impacted by the proposed power uprate do not result in significant adverse environmental impact. The Final Environmental Statement concluded that no significant environmental impact would result from operation of the Shearon Harris Nuclear Power Plant. This conclusion remains valid for the proposed SGR/power uprate. In accordance with the above evaluation, it can be concluded that no significant environmental impact will result from the proposed NSSS power level increase from 2787.4 MWt to NSSS power 2912.4 MWt.

2.26.4 References

1. Final Environmental Statement related to the operation of Shearon Harris Nuclear Power Plant, Units 1 and 2 (NUREG 0972) issued October 1983

2. Environmental Report Operating License Stage for the Shearon Harris Nuclear Power Plant, through Amendment 5.

3. Environmental Protection Plan, Appendix B to the Facility Operating License NPF-63.

4. NPDES Permit No. NC0039586

Table 2.26-1 ENVIRONMENTAL EVALUATION PARAMETERS

Parameter	Pre-Uprate Value	Uprate Value	Change (Uprate - Pre-Uprate)	FES Value	FES Table 4.1 Line No.
Cooling Tower Makeup Flow (gpm)	20,511 [1]	21,496 [1]	985	20,820 [3]	3
Circulating Water Flow (gpm)	483,000	487,600	4,600	483,000	6
Cooling Tower Evaporation [1] (gpm)	9,867 [2]	10,232[2]	365	10,170 [3]	5
Cooling Tower Drift (gpm)	10	10	negligible	10	FES Section 4.2.6.2
Cooling Tower Blowdown Temperature (°F) Blowdown temperature	71.6 (Jan.) 89.6 (July)	74.2 (Jan.)[4] 91.9 (July)[4]	2.6 (Jan.) 2.3 (July)	Not listed	Not listed
Cooling Tower blowdown [1] (gpm)	9,585 [1]	10,045 [1]	460	10,650 [3]	2

Notes for Table 2.26-1

1. Uprate and Pre-Uprate values based on 100% power, summer conditions (70°F wet bulb and 100% Relative Humidity) and cycles of Concentration of 2.1.

2. Based on annual average meteorology and operation at 100% power.

3. FES values for these continuous flows were based on an annual capacity factor of 85%. Refer to note "**" for FES Table 4.1

4. The values of Cooling Tower Basin temperature include a 2.2°F increase due to the measured Cooling Tower efficiency.