Westinghouse Non-Proprietary Class 3

WCAP - 15399

Carolina Power and Light Harris Nuclear Plant

Steam Generator Replacement/Uprating Analysis and Licensing Project

NSSS Licensing Report

September 2000

Westinghouse Electric Company LLC

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

NSSS LICENSING REPORT

WCAP-15399, WESTINGHOUSE NON-PROPRIETARY CLASS 3

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Harris Nuclear Plant
Steam Generator Replacement/Uprating
Analysis and Licensing Project
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ACRONYMS

AFWS	Auxiliary Feedwater System
AFW	Auxiliary Feedwater
ALARA	As Low as Reasonably Achievable
AMSAC	ATWS Mitigating System Actuation Circuitry
ANS	American National Standards
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
AVB	Anti-Vibration Bars
BAT	Boric Acid Tank
BATP	Boric Acid Transfer Pump
BIT	Boron Injection Tank
BOC	Beginning of Cycle
BOL	Beginning of Life
BOP	Balance of Plant
B&PV	Boiler and Pressure Vessel
BRS	Boron Recycle System
BTP	Branch Technical Position
BTRS	Boron Thermal Regeneration System
CAW	Customer Application for Withholding
CCW	Component Cooling Water
CCWS	Component Cooling Water System
C&FS	Condensate and Feedwater System
CG	Center of Gravity
CH/SI	Charging/Safety Injection
COMS	Cold Overpressure Mitigation System
CP&L	Carolina Power and Light
CRDM	Control Rod Drive Mechanism
CRSD	Core Radiation Source Data
CSA	Channel Statistical Allowance
CST	Condensate Storage Tank
CVCS	Chemical and Volume Control System
DE	Dose Equivalent
DEGB	Double-ended Guillotine Break
DEHL	Double Ended Hot Leg
DEPS	Double Ended Pump Suction
DF	Decontamination Factor

ACRONYMS (CONT.)

ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOC	End of Cycle
EOL	End of Life
EOP	Emergency Operating Procedure
EQ	Environmental Qualification
ERG	Emergency Response Guidelines
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
ESW	Emergency Service Water
FBCV	Feedwater Bypass Control Valve
FCV	Feedwater Control Valve
FDB	Flow Distribution Baffle
FIV	Feedwater Isolation Valve
FIV	Flow-Induced Vibration
FSAR	Final Safety Analysis Report
FW	Feedwater
FWLB	Feedwater Line Break
GDC	General Design Criteria
GDT	Gas Decay Tank
GFFDS	Gross Failed Fuel Detection System
GWPS	Gaseous Waste Processing System
HI	Heat Input
HHSI	High Head Safety Injection
HLSO	Hot-Leg Switchover
HNP	Harris Nuclear Plant
HX	Heat Exchanger
HZP	Hot Zero Power
IAM	Inputs, Assumptions and Methods
ICRP	International Commission on Radiation Protection
IFM	Intermediate Flow Mixing
ILRT	Integrated Leak Rate Test
IST	In-Service Testing
LBB	Leak Before Break
LHSI	Low-Head Safety Injection
LLC	Limited Liability Company

ACRONYMS (CONT.)

LOCA	Loss-of-Coolant Accident
LOOP	Loss-of-Offsite Power
LPP	Low Pressurizer Pressure
LPZ	Low Population Zone
LSGWL	Low-Low Steam Generator Water Level
LSP	Low Steamline Pressure
LTCC	Long-Term Core Cooling
LTOP	Low Temperature Over Pressurization
LTOPS	Low Temperature Over Pressurization System
LWPS	Liquid Waste Processing System
LWR	Light Water Reactor
M&E	Mass and Energy
MI	Mass Input
MFCV	Main Feedwater Control Valve
MFIV	Main Feedwater Isolation Valve
MMSBC	Mixed Mean Sump Boron Concentration
MPC	Maximum Permissible Concentrations
MS	Margin of Safety
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSS	Main Steam System
MSSV	Main Steam Safety Valve
NNS	Non-Nuclear Safety
NPSH	Net Positive Suction Head
NRS	Narrow-Range Span
NRC	Nuclear Regulatory Commission
NRL	Narrow Range Level
NSSS	Nuclear Steam Supply System
NUPPSCO	Nuclear Power Plant Standards Committee
OBE	Operating Basis Earthquake
OFA	Optimized Fuel Assembly
OPΔT	Over Power Delta Temperature
OTΔT	Over Temperature Delta Temperature
PCV	Pressure Control Valve
PCWG	Performance Capability Working Group
PI	Proportional and Integral
PLS	Precautions, Limitations and Setpoints

ACRONYMS (CONT.)

PORV	Power Operated Relief Valve
PRT	Pressurizer Relief Tank
PT	Pressure Transmitter
PWR	Pressurized Water Reactor
QMS	Quality Management System
RAI	Request for Additional Information
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
RHT	Recycle Holdup Tank
RPV	Reactor Pressure Vessel
RSG	Replacement Steam Generator
RTP	Rated Thermal Power
RTS	Reactor Trip System
RV	Reactor Vessel
RWST	Refueling Water Storage Tank
SB	Site Boundary
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SG	Steam Generator
SGBS	Steam Generator Blowdown System
SGLCS	Steam Generator Level Control System
SGR	Steam Generator Replacement
SGTP	Steam Generator Tube Plugging
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIS	Safety Injection System
SPC	Siemens Power Corporation
SRP	Standard Review Plan
SRSS	Square Root of the Sum of the Squares
SSDC	Systems Standard Design Criteria
SSE	Safe Shutdown Earthquake
SSPS	Solid State Protection System
SW	Service Water

ACRONYMS (CONT.)

TA	Total Allowance
TDAFW	Turbine Driven Auxiliary Feedwater
TDF	Thermal Design Flow
T/H	Thermal-Hydraulic
TID	Technical Information Document
TSP	Tube Support Plate
V5H	Vantage 5H
VCT	Volume Control Tank
WOG	Westinghouse Owners' Group

STEAM GENERATOR REPLACEMENT/UPRATING ANALYSIS AND LICENSING PROJECT

NSSS LICENSING REPORT

EXECUTIVE SUMMARY

PURPOSE

The Steam Generator Replacement (SGR)/Uprating Analysis and Licensing Project has been developed to support Carolina Power and Light (CP&L) in licensing the installation of replacement steam generators in the Harris Nuclear Plant (HNP) in conjunction with a power uprating. Implementation of the SGR/Uprating is planned for Refueling Outage No. 10, scheduled to begin on September 1, 2001.

In support of the SGR/Uprating Analysis and Licensing Project, Westinghouse has performed selected NSSS analyses and evaluations and provided the necessary documentation to support CP&L in their licensing application for HNP for the following SGR/Uprating conditions:

- Replacement of the existing HNP Model D4 steam generators with Model Δ75 steam generators, and

- An uprating of the NSSS power output from 2787.4 MWt to 2912.4 MWt (reactor core power of 2900 MWt).

The HNP SGR/Uprating Analysis and Licensing Project has been developed through the combined efforts of the project team members from CP&L, Siemens Power Corporation (SPC), Raytheon Engineers and Constructors, Bechtel Power Company, and Westinghouse Electric Company. Westinghouse is providing analysis, documentation and licensing support for the NSSS.

WESTINGHOUSE SCOPE FOR THE SGR/UPRATING PROJECT

In support of the HNP SGR/Uprating Project, Westinghouse has performed analyses and evaluations for the HNP Nuclear Steam Supply Systems (NSSS) to demonstrate that the NSSS is in compliance with applicable licensing criteria and requirements at the uprated NSSS thermal power of 2912.4 MWt (reactor core power of 2900 MWt). The scope of the Westinghouse analyses and evaluations included the NSSS performance parameters, design transients, systems, components, selected accidents, and selected analysis related to the nuclear fuel.

DOCUMENTATION

For the HNP SGR/Uprating Project, Westinghouse has drawn from previous experience performing the V. C. Summer Steam Generator Replacement program, in which Westinghouse Model Δ75 replacement steam generators were installed, and the J. M. Farley Uprate program. The engineering and licensing reports produced for the Farley Uprate program were used as guides for preparing the HNP engineering and licensing documentation. Moreover, the Farley Licensing Report, together with NRC review and comment resolution documentation, has been used as a benchmark for the HNP SGR/Uprating Licensing Report. Accordingly, the HNP Licensing Report includes applicable responses based on the NRC Requests for Additional Information (RAI) issued for the Farley Uprate program.

This NSSS Licensing Report provides a description of the NSSS analyses and evaluations performed by Westinghouse for the SGR/Uprating project.

A description of the analyses and evaluations performed for the Balance of Plant (BOP) secondary systems and components is provided by Bechtel, SPC, and Raytheon in a separate BOP Licensing Report.

CONCLUSION

The results of the Westinghouse NSSS analyses and evaluations demonstrate that the applicable licensing criteria and requirements are satisfied for the NSSS performance parameters, design transients, systems, components, and selected accidents with the replacement steam generators at the uprated power conditions. The results also show that operation of HNP with the replacement steam generators at the current NSSS power is bounded by the analyses and evaluations performed at the uprated conditions.

1.0 INTRODUCTION

1.1 Purpose and Scope

The purpose of the SGR/Uprating project is to provide the engineering analyses and evaluations, licensing efforts, and documentation necessary to support licensing the Harris Nuclear Plant with:

- Model Δ75 steam generators replacing the existing Model D4 steam generators, and
- An uprating of the NSSS power output from 2787.4 MWt to 2912.4 MWt, with a corresponding core power of 2900 MWt.

This NSSS Licensing Report documents the results of analyses and evaluations performed in support of the Harris Nuclear Plant (HNP) Steam Generator Replacement and Power Uprating (SGR/Uprating) Analysis and Licensing Project. The analyses and evaluations performed by Westinghouse are based on Westinghouse Model Δ75 replacement steam generators (RSGs) operating at the following conditions:

- Uprated power level of 2912.4 MWt (2900 MWt core power)
- Average temperature range of 572°F–588°F
- Full-power feedwater temperature of 375°F–440°F
- Steam generator tube plugging (SGTP) level of 0 to 10 percent
- Siemens Power Corporation (SPC) fuel, unless otherwise noted

The following subsections of the Westinghouse Engineering Report are not Westinghouse scope and are not included in this report. These sections will be provided separately to CP&L by others:

Analysis by Others	ER Section Number
Reactor Vessel Integrity	5.1.2
Fuel Assemblies	5.3
Loss of Coolant Accident (LOCA) Transients	
Large Break LOCA	6.1.1
Small Break LOCA	6.1.2
Rod Ejection Accident Analysis	6.1.5
Non-LOCA Transients	6.2

Analysis by Others	ER Section Number
Nuclear Fuel	
Core Thermal Hydraulic Design	7.1
Fuel Core Design	7.2
Fuel Rod Design and Performance	7.3
Neutron Fluence	7.5

1.2 Methodology and Acceptance Criteria

The Harris Nuclear Plant SGR/Uprating project has been structured consistent with the methodology established in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," dated 1983. Since its submittal to the NRC, the methodology has been used successfully as a basis for power uprate projects on over 20 pressurized water reactors (PWRs), one of the latest being the Farley Uprate Power Project in 1998.

The methodology in WCAP-10263 established the ground rules and criteria for power uprate projects, including the broad categories that must be addressed, such as NSSS performance parameters, design transients, systems, components, accidents and nuclear fuel as well as the interfaces between the NSSS and the BOP fluid systems. Inherent in this methodology are key points that promote correctness, consistency, and licensability. The key points include the use of well-defined analysis input assumptions/parameter values, use of currently approved analytical techniques (e.g., methodologies and computer codes), and use of currently applicable licensing criteria and standards.

For each analysis or evaluation, a set of inputs, assumptions and methods (IAMs) were submitted for CP&L's approval prior to starting the evaluation. This process was used to identify and control the input assumptions for the SGR/Uprating analysis.

SGR/Uprating analyses were performed to reflect the as-built and as-operated plant. If plant drawings (as built) and/or plant documentation (site-specific procedures) were required in order to obtain the latest plant information for use in SGR/Uprating analyses, they were obtained from CP&L and used as appropriate to obtain the needed information.

The SGR/Uprating analyses and evaluations were performed in accordance with Westinghouse quality assurance requirements defined in the Westinghouse Quality Management System (QMS) procedures, which comply with 10 CFR 50 Appendix B criteria. These analyses and evaluations are in conformance with Westinghouse and industry codes, standards, and regulatory requirements applicable to HNP. Assumptions and acceptance criteria are provided in the appropriate sections of this report.

1.2.1 Analytical Computer Codes

The SGR/Uprating program analyses and evaluations were performed using currently approved analytical techniques to demonstrate compliance with the licensing criteria and standards that apply to HNP.

In performing the SGR/Uprating analyses, the Westinghouse analysts used methodologies and principle computer codes that are currently approved by the Nuclear Regulatory Commission (NRC). The NRC-approved techniques are the same as those used for current HNP analyses and described in the HNP Final Safety Analysis Report, except for the following areas where NRC approved methodology were applied to HNP for the first time:

- Main Steamline Break (MSLB) mass and energy (M&E) releases analysis for outside containment (Section 6.5.2) used the 1979 ANS-5.1 Standard Decay Model. This decay heat model has been previously used by Westinghouse in other analyses for HNP including the MSLB M&E releases inside containment and has been approved by the NRC. The revised HNP analysis was performed with the approved Westinghouse model for inside and outside containment of WCAP-8822, with Supplement 1 and WCAP 10961.

- The Steam Generator Tube Rupture (SGTR) radiological consequences (Section 6.3.3) analysis was updated to include the following methodologies and significant input assumption changes.

  - Use of the latest resolution of the steam generator tube uncovery issue as documented in WCAP-13247

  - Transition to the use of International Commission on Radiological Protection ICRP-30 conversion factors

  - Reduction of the assumed coolant activity from 1.0 μCi/g to 0.35 μCi/g.

  - Not taking credit for the assumed scrubbing of the flashed break flow

  - The inclusion of iodine contribution to the development of the total whole body dose

Table 1-1 provides a list of the principal analysis computer codes used for the SGR/Uprating analysis. The table delineates which codes have been previously used and approved for HNP.

1.2.2 Westinghouse Proprietary Information Designations

There is information contained in this report that is considered Westinghouse Proprietary. The specific information is contained within the brackets with designated superscripted letters (a, b, and c).

Since this report contains information proprietary to Westinghouse Electric Company LLC, the information must be withheld from public disclosure by the Nuclear Regulatory Commission (NRC) when it is reviewing the documentation. Westinghouse requests the NRC to protect the bracketed information from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information submitted to the NRC, the information that is proprietary (in the proprietary versions of documentation) is contained within brackets. The information that was contained within the brackets in the proprietary versions is deleted in the non-proprietary version. The justification for claiming the information designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence. The description of the superscripts is included in the Westinghouse Customer Application for Withholding (CAW) transmittal that would accompany the Westinghouse report when it is submitted to the NRC.

1.3 Technical Basis for Significant Hazards Evaluation

This NSSS Licensing Report provides the technical basis for the No Significant Hazards Evaluation of the proposed Technical Specification changes for the HNP SGR/Uprating project.

1.4 Conclusions

The results of the NSSS analyses and evaluations demonstrate that the HNP NSSS is in compliance with applicable licensing criteria and requirements and can operate acceptably with the RSGs at the uprated power of 2912.4 MWt, or at the current NSSS power of 2787.4 MWt.

The following licensing scope deliverables are provided to CP&L under separate cover:

- Safety Evaluation (10CFR50.59)
- No Significant Hazards Evaluation (10CFR50.92)
- Proposed HNP Technical Specification Markups
- Proposed HNP Final Safety Analysis Report Markups

Table 1-1 HNP SGR/Uprating Analysis And Licensing Project Westinghouse Principle Computer Codes			
Report Section	**Analysis**	**Computer Code[1]**	**Previously Used on HNP**
4.3	NSSS Control Systems	LOFT12, LOFT4	Yes
5.2	Reactor Internals Analysis	THRIVE WECAN MULTIFLEX ANSYS	Yes Yes Yes No[2]
5.5.1 and 5.5.3	Reactor Coolant Loop Piping/Supports and Class 1 Auxiliary Lines	WECAN WESTDYN STRUDL	Yes
6.3.1	SGTR - Margin to SG Overfill	LOFTTR2	Yes
6.3.2	SGTR - T-H Analysis for Radiological Consequences	LOFTTR2	Yes
6.4	LOCA M&E Releases	SATAN VI WREFLOOD FROTH EPITOME	Yes
6.5	MSLB M&E Releases	LOFTRAN	Yes
6.6	LOCA Hydraulic Forces	MULTIFLEX LATFORC FORCE2 THRUST	Yes
6.8	Anticipated Transient Without Scram	LOFTRAN NOTRUMP	Yes Yes
7.4	Heat Generation Rates	DORT/SORCERY	Yes
7.6	Source Terms	ORIGEN2.1	Yes

Notes:

1. All codes listed are maintained under Westinghouse Configuration Control.
2. This is the first application of the ANSYS code at HNP. This is an industry standard code for finite element modeling.

2.0 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) PARAMETERS

As part of the SGR/Uprating Project for HNP, NSSS performance analyses were performed to develop NSSS design parameters. Bounding NSSS Performance Capability Working Group (PCWG) parameters were developed for use in the analyses and evaluations of NSSS design transients, systems, and components.

2.1 Design PCWG Parameters and Additional Cases

2.1.1 Introduction and Background

The NSSS PCWG parameters are the fundamental parameters that are used as primary input in NSSS analyses. They provide the Reactor Coolant System (RCS) and secondary system conditions (temperatures, pressures, flow) that are used as the basis for the analyses and evaluations of design transients, systems, components, and accident analyses in the Westinghouse scope (M&E Releases, Source Terms, Steam Generator Tube Rupture, Anticipated Transients without Scram, etc.).

The design PCWG parameters are established using conservative assumptions in order to provide bounding conditions appropriate for the NSSS analyses. For example, the assumed RCS flow is the Thermal Design Flow (TDF), which is a conservatively low flow that accounts for flow measurement uncertainty and bounds the maximum expected steam generator tube plugging (SGTP) level.

The design PCWG parameters are developed to enable CP&L to have operating flexibility by using a range of conditions based on the vessel average temperature (T_{avg}), the feedwater temperature and the SGTP level. As shown in Table 2-1, the eight PCWG cases specify a T_{avg} range between 572.0°F and 588.8°F, a feedwater temperature range between 440°F and 375°F, and a SGTP level between 0 to 10 percent.

Westinghouse developed a set of design PCWG parameters for those analyses or evaluations performed at the current NSSS power operation of 2787.4 MWt and the uprated NSSS power operation of 2912.4 MWt for the SGR/Uprating Project. Westinghouse also developed part power cases for the Turbine-Generator evaluations.

2.1.2 Description of Analyses and Evaluations

The calculation of the PCWG parameters established for the current full power operation are summarized in Table 2-2 and for the uprated full power operation in Table 2-3. Listed below are the major inputs used in the generation of the PCWG parameters.

- The current NSSS power level is set at 2787.4 MWt (2775 MWt core).

- The uprated NSSS power level is set at 2912.4 MWt (2900 MWt core).

- Siemens Fuel is incorporated into the SGR/Uprating analysis.

- The TDF of 92,600 gpm/loop incorporated sufficient margin to support 10 percent average SGTP. This flow was applied for all eight design cases, even those which assumed zero percent SGTP, in order to be consistent and for conservatism. Both 0-percent and 10-percent SGTP is assumed.

- Design core bypass flow is assumed to be 7.1 percent.

- A range of full power normal operating T_{avg} from 572.0°F to 588.8°F is selected for the analyses.

- A range of full-power feedwater temperature of 375°F to 440°F is selected for the analyses.

2.1.2.1 Current Power Cases

A total of eight design cases were developed, according to Table 2-1, using the current NSSS power of 2787.4 MWt.

2.1.2.2 Uprated Power Cases

A total of eight design basis cases were developed according to Table 2-1 for the power uprating. These PCWG-approved cases form the design basis for the SGR/Uprating Project.

Westinghouse also provided 32 additional parameter cases (based on HNP Technical Specification minimum and maximum RCS flow conditions and best estimate RCS flow at feedwater temperatures of 375.0°F, 435.0°F, and 440.0°F and SGTP levels of 0, 3, and 10 percent) for use by CP&L in fuel related analyses and accidents and in predicting actual plant operating conditions. Westinghouse evaluated the 32 additional cases to determine whether they provided a more conservative basis for their NSSS design bases analysis. In the cases where they did, the more conservative best estimate parameters were used as conservative input assumptions in the NSSS design bases analysis.

2.1.2.3 Part Power Conditions

The part power cases were developed to support evaluation of the turbine-generator by CP&L. Cases were provided for Δ75 Steam Generators at power levels of 100 percent, 75 percent, 50 percent, and 25 percent of 2912.4 MWt.

2.1.3 Acceptance Criteria

The primary acceptance criteria for the determination of the design PCWG parameters are: (1) the parameters must provide CP&L with adequate flexibility and margin for plant operation, and (2) the parameters must maintain current thermal design flow and minimum measured flow requirements (shown in Tables 2-4 and 2-5) based on the assumed SGTP level.

2.1.4 Results

Table 2-4 provides the eight design NSSS PCWG parameter cases that are used as the basis for the NSSS design bases analyses or evaluations at the current power conditions.

Table 2-5 provides the eight design NSSS PCWG parameter cases that are used as the basis for the NSSS design bases analyses or evaluations at the uprated power conditions.

Table 2-6 provides the part power cases developed at power levels of 100 percent, 75 percent, 50 percent, and 25 percent of 2912.4 MWt.

2.1.5 Conclusions

The PCWG current and uprated power cases were issued to the various Westinghouse engineering organizations. The Westinghouse analysts performed their analyses and evaluations based on the parameter set or sets that were most limiting so that the analyses would support operation of HNP over the range of conditions specified.

Table 2-1 NSSS PCWG Cases at Current Power and Uprated Power			
Case No.	**SGTP (%)**	**Average Temp. (°F)**	**Feedwater Temp. (°F)**
1	0	572.0	375
2	0	588.8	375
3	10	572.0	375
4	10	588.8	375
5	0	572.0	440
6	0	588.8	440
7	10	572.0	440
8	10	588.8	440

Table 2-2 NSSS PCWG Input Parameters at Current Power Level	
Parameter	**Value**
Fuel Type and Features	
Type	Siemens
IFMs (Yes/No)	Yes
Fuel Rod OD (in.)	0.376
NSSS Thermal Power	
NSSS Power (MWt)	2787.4
Core Power (MWt)	2775
Net Heat Input (MWt)	12.4
RCS Flow	
Thermal Design Flow (gpm/loop)	92,600
Total Core Bypass Flow (%)	7.1
Thimble Plugs (In/Out)	In
RCS Temperature	
Vessel Average (°F)	572.0, 588.8
RCS Pressure (psia)	2250
Steam Generator (Model Δ75)	
Steam Generator Tube Plugging Level, Min/Max (%)	0 / 10
Main Feedwater Temperatures (°F)	375, 440*

* Temperature range specified for the SGR/Uprating project by CP&L and is consistent with the range for the uprated parameter cases.

Table 2-3 NSSS PCWG Input Parameters At Uprated Power Level	
Parameter	
Fuel Type and Features	
Type	Siemens
IFMs (Yes/No)	Yes
Fuel Rod OD (in.)	0.376
NSSS Thermal Power	
NSSS Power (MWt)	2912.4
Core Power (MWt)	2900
Net Heat Input (MWt)	12.4
RCS Flow	
Thermal Design Flow (gpm/loop)	92,600
Total Core Bypass Flow (%)	7.1
Thimble Plugs (In/Out)	In
RCS Temperature	
Vessel Average (°F)	572.0, 588.8
RCS Pressure (psia)	2250
Steam Generator (Model Δ75)	
Steam Generator Tube Plugging Level, Min/Max (%)	0 / 10
Main Feedwater Temperatures (°F)	375, 440

Table 2-4
NSSS PCWG Parameters at HNP Current Power Conditions

THERMAL DESIGN PARAMETERS	Original Basis(1)	Case 1(2)	Case 2(2)	Case 3(2)	Case 4(2)
NSSS Power, %	100	100	100	100	100
MWt	2785	2787.4	2787.4	2787.4	2787.4
10^6 BTU/hr	9503	9511	9511	9511	9511
Reactor Power, MWt	2775	2775	2775	2775	2775
10^6 BTU/hr	9469	9469	9469	9469	9469
Thermal Design Flow, Loop gpm	97,600	92,600	92,600	92,600	92,600
Reactor 10^6 lb/hr	109.0	106.1	103.6	106.1	103.6
Reactor Coolant Pressure, psia	2250	2250	2250	2250	2250
Core Bypass, %	6.1	7.1	7.1	7.1	7.1
Reactor Coolant Temperature, °F					
Core Outlet	623.8	610.6	626.2	610.6	626.2
Vessel Outlet	620.2	605.9	621.8	605.9	621.8
Core Average	592.5	576.1	593.2	576.1	593.2
Vessel Average	588.8	572.0	588.8	572.0	588.8
Vessel/Core Inlet	557.4	538.1	555.8	538.1	555.8
Steam Generator Outlet	557.1	537.8	555.5	537.8	555.5
Steam Generator					
Steam Temperature, °F	540.2	525.6	543.8	523.6	541.8
Steam Pressure, psia	964	852	993	838	978
Steam Flow, 10^6 lb/hr total	12.20	11.21	11.28	11.21	11.27
Feed Temperature, °F	435	375	375	375	375
Moisture, % max.	0.25	0.10	0.10	0.10	0.10
Steam Generator Tube Plugging, %	0	0	0	10	10
Zero Load Temperature, °F	557	557	557	557	557
HYDRAULIC DESIGN PARAMETERS					
Pump Design Point, Flow (gpm)/Head (ft.)	103,400/276				
Best Estimate Flow, gpm		102,200	102,200	100,000	100,000
Mechanical Design Flow, gpm	107,100				
Minimum Measured Flow, gpm total		293,540	293,540	293,540	293,540

FOOTNOTES:
(1) Original parameters reflect operation with a model D4 steam generator.
(2) Parameters assume operation with a model Δ75 steam generator.

Table 2-4 (continued)
NSSS PCWG Parameters at HNP Current Power Conditions

THERMAL DESIGN PARAMETERS	Original Basis(1)	Case 5(2)	Case 6(2)	Case 7(2)	Case 8(2)
NSSS Power, %	100	100	100	100	100
MWt	2785	2787.4	2787.4	2787.4	2787.4
10^6 BTU/hr	9503	9511	9511	9511	9511
Reactor Power, MWt	2775	2775	2775	2775	2775
10^6 BTU/hr	9469	9469	9469	9469	9469
Thermal Design Flow, Loop gpm	97,600	92,600	92,600	92,600	92,600
Reactor 10^6 lb/hr	109.0	106.1	103.6	106.1	103.6
Reactor Coolant Pressure, psia	2250	2250	2250	2250	2250
Core Bypass, %	6.1	7.1	7.1	7.1	7.1
Reactor Coolant Temperature, °F					
Core Outlet	623.8	610.6	626.2	610.6	626.2
Vessel Outlet	620.2	605.9	621.8	605.9	621.8
Core Average	592.5	576.1	593.2	576.1	593.2
Vessel Average	588.8	572.0	588.8	572.0	588.8
Vessel/Core Inlet	557.4	538.1	555.8	538.1	555.8
Steam Generator Outlet	557.1	537.8	555.5	537.8	555.5
Steam Generator					
Steam Temperature, °F	540.2	525.6	543.8	523.6	541.8
Steam Pressure, psia	964	852	993	838	978
Steam Flow, 10^6 lb/hr total	12.20	12.23	12.30	12.22	12.29
Feed Temperature, °F	435	440	440	440	440
Moisture, % max.	0.25	0.10	0.10	0.10	0.10
Steam Generator Tube Plugging, %	0	0	0	10	10
Zero Load Temperature, °F	557	557	557	557	557
HYDRAULIC DESIGN PARAMETERS					
Pump Design Point, Flow (gpm)/Head (ft.)	103,400/276				
Best Estimate Flow, gpm		102,200	102,200	100,000	100,000
Mechanical Design Flow, gpm	107,100				
Minimum Measured Flow, gpm total		293,540	293,540	293,540	293,540

FOOTNOTES:
(1) Original parameters reflect operation with a model D4 steam generator.
(2) Parameters assume operation with a model Δ75 steam generator.

Table 2-5
NSSS PCWG Parameters at HNP Uprated Power Conditions

THERMAL DESIGN PARAMETERS	Original Basis(1)	Case 1(2)	Case 2(2)	Case 3(2)	Case 4(2)
NSSS Power, %	100	104.6	104.6	104.6	104.6
MWt	2785	2912.4	2912.4	2912.4	2912.4
10^6 BTU/hr	9503	9938	9938	9938	9938
Reactor Power, MWt	2775	2900	2900	2900	2900
10^6 BTU/hr	9469	9895	9895	9895	9895
Thermal Design Flow, Loop gpm	97600	92600	92600	92600	92600
Reactor 10^6 lb/hr	109.0	106.3	103.8	106.3	103.8
Reactor Coolant Pressure, psia	2250	2250	2250	2250	2250
Core Bypass, %	6.1	7.1	7.1	7.1	7.1
Reactor Coolant Temperature, °F					
Core Outlet	623.8	612.2	627.7	612.2	627.7
Vessel Outlet	620.2	607.4	623.2	607.4	623.2
Core Average	592.5	576.4	593.4	576.4	593.4
Vessel Average	588.8	572.0	588.8	572.0	588.8
Vessel/Core Inlet	557.4	536.6	554.4	536.6	554.4
Steam Generator Outlet	557.1	536.3	554.1	536.3	554.1
Steam Generator					
Steam Temperature, °F	540.2	523.7	541.9	521.6	539.9
Steam Pressure, psia	964	839	978	824	962
Steam Flow, 10^6 lb/hr total	12.20	11.71	11.78	11.71	11.77
Feed Temperature, °F	435	375	375	375	375
Moisture, % max.	0.25	0.10	0.10	0.10	0.10
Steam Generator Tube Plugging, %	0	0	0	10	10
Zero Load Temperature, °F	557	557	557	557	557
HYDRAULIC DESIGN PARAMETERS					
Pump Design Point, Flow (gpm)/Head (ft.)	103,400/276				
Best Estimate Flow, gpm		102,200	102,200	100,000	100,000
Mechanical Design Flow, gpm	107,100				
Minimum Measured Flow, gpm total		293,540	293,540	293,540	293,540

FOOTNOTES:
(1) Original parameters reflect operation with a model D4 steam generator.
(2) Parameters assume operation with a model Δ75 steam generator.

Table 2-5 (continued)
NSSS PCWG Parameters at HNP Uprated Power Conditions

THERMAL DESIGN PARAMETERS	Original Basis(1)	Case 5(2)	Case 6(2)	Case 7(2)	Case 8(2)
NSSS Power, %	100	104.6	104.6	104.6	104.6
MWt	2785	2912.4	2912.4	2912.4	2912.4
10^6 BTU/hr	9503	9938	9938	9938	9938
Reactor Power, MWt	2775	2900	2900	2900	2900
10^6 BTU/hr	9469	9895	9895	9895	9895
Thermal Design Flow, Loop gpm	97600	92600	92600	92600	92600
Reactor 10^6 lb/hr	109.0	106.3	103.8	106.3	103.8
Reactor Coolant Pressure, psia	2250	2250	2250	2250	2250
Core Bypass, %	6.1	7.1	7.1	7.1	7.1
Reactor Coolant Temperature, °F					
Core Outlet	623.8	612.2	627.7	612.2	627.7
Vessel Outlet	620.2	607.4	623.2	607.4	623.2
Core Average	592.5	576.4	593.4	576.4	593.4
Vessel Average	588.8	572.0	588.8	572.0	588.8
Vessel/Core Inlet	557.4	536.6	554.4	536.6	554.4
Steam Generator Outlet	557.1	536.3	554.1	536.3	554.1
Steam Generator					
Steam Temperature, °F	540.2	523.7	541.9	521.6	539.9
Steam Pressure, psia	964	839	978	824	962
Steam Flow, 10^6 lb/hr total	12.20	12.77	12.85	12.76	12.84
Feed Temperature, °F	435	440	440	440	440
Moisture, % max.	0.25	0.10	0.10	0.10	0.10
Steam Generator Tube Plugging, %	0	0	0	10	10
Zero Load Temperature, °F	557	557	557	557	557
HYDRAULIC DESIGN PARAMETERS					
Pump Design Point, Flow (gpm)/Head (ft.)	103,400/276				
Best Estimate Flow, gpm		102,200	102,200	100,000	100,000
Mechanical Design Flow, gpm	107,100				
Minimum Measured Flow, gpm total		293,540	293,540	293,540	293,540

FOOTNOTES:

(1) Original parameters reflect operation with a model D4 steam generator.
(2) Parameters assume operation with a model Δ75 steam generator.

Table 2-6

Summary of Turbine Evaluation Points

For SGR/Uprating Program

Parameter	100% of 2912.4 MWt	75% of 2912.4 MWt	50% of 2912.4 MWt	25% of 2912.4 MWt
NSSS Power (MWt)	2912.4	2184.3	1456.2	728.1
Vessel High T_{avg} (°F)	588.8 @ 0% SGTP	580.8 @ 0% SGTP	572.9 @ 0% SGTP	564.9 @ 0% SGTP
Vessel Low T_{avg} (°F)	572 @ 10% SGTP	568.2 @ 10% SGTP	564.5 @ 10% SGTP	560.7 @ 10% SGTP
Best Estimate Flow Rate, Loop (gpm)	102,200 @ 0% SGTP	102,200 @ 0% SGTP	102,200 @ 0% SGTP	102,200 @ 0% SGTP
Best Estimate Flow Rate, Loop (gpm)	100,000 @ 10% SGTP	100,000 @ 10% SGTP	100,000 @ 10% SGTP	100,000 @ 10% SGTP
Feedwater Temp. (°F)	440	406	369	320
Steam Moisture Content (%)	0.10	0.10	0.10	0.10

3.0 NSSS AND AUXILIARY EQUIPMENT DESIGN TRANSIENTS

This section discusses the generation of NSSS and Auxiliary Equipment Design Transients for the SGR/Uprating conditions. The design transients were reviewed and revised as necessary, for use at the HNP SGR/Uprating conditions. The revised NSSS design transient curves were provided to all system and component designers for use in their specific analyses. Section 3.1 describes the evaluation performed. Auxiliary equipment design transients were also evaluated to determine whether they remain applicable for use in the SGR/Uprating analysis of all the auxiliary equipment in the NSSS. The results of this evaluation are presented in Section 3.2.

3.1 NSSS Design Transients

3.1.1 Introduction and Background

As part of the original design and analyses of the NSSS components for the HNP, NSSS design transients (i.e., temperature and pressure transients) were specified (Reference 1) for use in the analyses of the cyclic behavior of the NSSS components. To provide the necessary high degree of integrity for the NSSS components, the transient parameters selected for component fatigue analyses are based on conservative estimates of the magnitude and frequency of the temperature and pressure transients resulting from various plant operating conditions. The transients selected for use in component fatigue analyses are representative of operating conditions that would be considered to occur during plant operations of possible significance to component cyclic behavior due to their severity or frequency. The selected transients are representative of plant transients which, when used as a basis for component fatigue analysis, would provide confidence that the component is appropriate for its application over the operating license period of the plant. For purposes of analysis, the number of transient occurrences is based on an operating license period of 40 years.

3.1.2 Description of Analyses and Evaluations

Initial plant conditions for the NSSS design transients are based primarily on the NSSS Performance Capability Working Group (PCWG) parameters as discussed in Section 2 of this report.

The NSSS design transients for the SGR/Uprating program and the original HNP design transients (Reference 1) were compared, and any differences were evaluated. The effect of the following changes on the NSSS design transients were also evaluated:

- Main steam safety valves blowdown
- Moisture separator designs for the HNP Model Δ75 replacement steam generator

The latest designs of the HNP Δ75 moisture separator have resulted in a change in the steam generator level control setpoint and steam generator mass. Additionally, the steam generator thermal-hydraulic characteristics have changed.

3.1.3 Acceptance Criteria and Results

The NSSS design transients are input to the primary and secondary component structural and fatigue analysis/evaluations. The final acceptance is determined by the component stress and fatigue analyses presented in Section 5.

Consistent with the original NSSS design transients, the NSSS design transients for the SGR/Uprating are conservative representations of transients which, when used as a basis for component fatigue analyses, provide confidence that the component is appropriate for its application over the operating license period of the plant. Also, consistent with the original NSSS design transients, the number of transient occurrences are based on an operating license period of 40 years.

3.1.4 Conclusions

The NSSS design transients for the Δ75 replacement steam generators at the uprated power level are also applicable for the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

3.1.5 Reference

1. Systems Standard Design Criteria (SSDC) 1.3, Rev. 2, April 15, 1974.

3.2 Auxiliary Equipment Design Transients

3.2.1 Introduction

As part of the original design and analysis of the NSSS auxiliary components (i.e., NSSS auxiliary pumps, valves, and heat exchangers) for the Harris Nuclear Plant (HNP), auxiliary equipment design transients (i.e., temperature and pressure transients) were specified for use in the analyses of the cyclic behavior of the NSSS auxiliary components. To provide the necessary high degree of integrity for the NSSS auxiliary components, the transient design parameters selected for component fatigue analyses were based on conservative estimates of the magnitude and frequency of the temperature and pressure transients resulting from various postulated plant operating conditions. The transients selected for use in component fatigue analyses are representative operating conditions that are considered to be sufficiently severe or frequent as to impact component cyclic behavior. The transients are selected to be sufficiently conservative such that, when used as a basis for component fatigue analysis, they provide confidence that the component will perform as intended over the operating license period of the plant. For purposes of analysis, the number of transient occurrences was based on an operating license period of 40 years.

3.2.2 Input Parameters and Assumptions

The review of the NSSS auxiliary equipment design transients was based on a comparison between the NSSS PCWG parameters for the Steam Generator Replacement/Uprating (SGR/Uprating) as discussed in Chapter 2 of this report and the design parameters that make up the current auxiliary equipment design transients. The current transients that are specified for HNP are contained in Reference 1.

3.2.3 Description of Analyses and Evaluation

A review of the current auxiliary equipment transients determined that the only transients that could be impacted by the SGR/Uprating are those temperature transients that are impacted by full-load NSSS operating temperatures, namely T_{hot} and T_{cold}. These transients are currently based on an assumed full-load NSSS worst case T_{hot} of 630°F and worst case T_{cold} of 560°F. These NSSS temperatures were originally selected to ensure that the resulting design transients would be conservative for a wide range of NSSS operating temperatures.

3.2.4 Acceptance Criteria and Results

A comparison of the SGR/Uprating NSSS PCWG parameter ranges for T_{hot} (604.2° to 623.2°F) and T_{cold} (536.6° to 557.4°F) with the T_{hot} and T_{cold} values used to develop the current design transients indicates that the SGR/ Uprating T_{hot} and T_{cold} parameter ranges are lower. Therefore, the actual temperature transients (that is, the change in temperature from T_{hot} or T_{cold} dictated by

the SGR/Uprating parameters to a lower auxiliary system-related temperature or visa versa) are less severe than the current design temperature transients.

3.2.5 Conclusions

As the temperature transients dictated by the SGR/Uprating conditions are less limiting than those that established the current auxiliary equipment design transients, it can be concluded that all the applicable auxiliary equipment design transients for HNP still apply for the SGR/Uprating conditions.

The results of the evaluation are consistent with and continue to comply with the current HNP licensing basis acceptance requirements. The results obtained with the Δ75 replacement steam generators at the uprated NSSS power of 2912.4 MWt bound plant operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

3.2.6 References

1. Westinghouse Systems Standard Design Criteria (SSDC) document 1.3, Rev. 2, April 15, 1974.

4.0 NSSS SYSTEMS

This section presents the results of the analyses and evaluations performed in the NSSS systems area to support the HNP SGR/Uprating. The systems addressed in this section include fluid systems, NSSS/BOP interface systems, and control systems. The detailed results and conclusions of each analysis/evaluation are presented within each subsection.

4.1 NSSS Fluid Systems

The fluid systems evaluations described in this section were performed at the systems level. For the fluid systems supplied for HNP by Westinghouse, the applicable system functional requirements and performance criteria were reviewed to ensure that each system remains capable of performing its design basis functions at the SGR/Uprating conditions. A justification is provided for systems that are judged to be unaffected by the uprated plant conditions.

4.1.1 Reactor Coolant System

4.1.1.1 Introduction

The RCS consists of three heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump (RCP), which circulates the water through the loops and reactor vessel, and a steam generator (SG), where heat is transferred to the Main Steam System (MSS). In addition, the RCS contains a pressurizer which controls RCS pressure through electrical heaters, water sprays, power-operated relief valves (PORVs) and spring-loaded safety/relief valves. The steam discharged from the PORVs and safety/relief valves flows through interconnecting piping to the pressurizer relief tank (PRT).

This section identifies the key functions of the RCS and identifies which functions are potentially impacted by the SGR/Uprating program.

The key RCS functions are as follows:

- The RCS transfers heat generated in the reactor core to the MSS via the SGs.
- When the core is subcritical and RCS temperatures are approximately 350°F and lower, the RCS provides means to transfer decay and sensible heat to the RHRS.
- The RCS fluid acts as a moderator of neutrons by slowing the neutrons to lower thermal energy states and increasing the probability of thermal fission.
- The RCS fluid is a solvent and carrier of boric acid, used as a neutron poison.

- The RCS is the second of three "barriers" against fission product release to the environment. (The fuel cladding is the first barrier, and the containment building is the third.)

- The RCS provides means for pressure control via use of pressurizer heaters and spray flow.

The calculated RCS design operating conditions at the uprated power are presented in Section 2 of this report. The primary changes in NSSS Performance Capability Working Group (PCWG) parameters that impact the RCS functions include the increase in core power, the allowable operating range for average RCS temperature (T_{avg}), and reduced thermal design flow.

The potential impact of the SGR/Uprating conditions on the previous RCS functions are described below.

- The core power increase will affect the total amount of heat transferred to the MSS. Verification that the major components can support this normal heat removal function is addressed in Section 5 of this report.

- During the second phase of plant cooldown, the RHRS will be required to remove larger amounts of decay heat from the RCS. Section 4.1.4 of this report addresses the RHRS cooldown capability at uprated conditions.

- Due to the replacement steam generator, the RCS volume will be affected, which will in turn affect the amount of boron required for reactivity control.

- The RCS heat capacity of the primary coolant system will be affected by the Model $\Delta 75$ steam generators that will be installed.

- The pressurizer spray flow rate was affected by changes in the reactor coolant best estimate flow rate, which depends on the steam generator tube plugging level and the associated effect on the pressure drop across the reactor vessel.

4.1.1.2 RCS Loop Operating Pressures

4.1.1.2.1 Introduction

This evaluation was performed to determine if the HNP RCS would undergo any significant pressure changes due to variations in flow and temperature resulting from the SGR/Uprating. The analysis calculated the operating pressures at the SGR/Uprating plant conditions.

The objective of this evaluation was to calculate the RCS pressure at various locations within the RCL at steady-state conditions. These results, coupled with the RCS pressure deviations determined by NSSS design transients and Low Temperature Over Pressurization (LTOP)

analyses, predict the pressure for a range of conditions. This determines the sufficiency of the RCS and associated Class 1 components.

4.1.1.2.2 Description of Analyses and Evaluations

The RCS loop pressures were determined based on hydraulic resistances and pump data. Actual component elevations were used as input to account for elevation effects. The correct pressurizer pressure and fluid density were used as input for the liquid space and surge line to ensure that the correct static pressure was imposed on the hot leg. A fluid density corresponding to cold-leg conditions was used for the RCS loops and vessel. This was done so that the unrecoverable pressure drops would be correctly calculated based on the NSSS hydraulic resistances.

The operating pressure was calculated for the following RCS main-coolant-loop locations:

- Pressurizer Surge Line Connection to Loop 2 Hot Leg
- Reactor Vessel Inlet
- Reactor Vessel Outlet
- Steam Generator Inlet
- Steam Generator Outlet
- Reactor Coolant Pump Inlet
- Reactor Coolant Pump Outlet

Because the variations in the operating pressure are not expected to be great over large segments of the RCS loops (e.g., cold leg, hot leg, etc.), all primary components and connection points to other Class 1 equipment and piping will be bounded by the calculated operating pressures at the locations specified above.

4.1.1.2.3 Acceptance Criteria

For the SGR/Uprating, the operating pressures for the RCS loop cannot exceed the design pressures of the primary loop components and piping.

4.1.1.2.4 Results

The operating pressure was calculated for each location for full-power operation at high and low RCS temperatures. The results demonstrate that at full power the temperature affects the pressure only slightly. However, the pressure in the RCS loop for best estimate flow is shown to be lower than 2300 psig for full-power operation at any temperature.

4.1.1.2.5 Conclusions

This analysis demonstrated that under normal operating conditions the pressure at any location in the RCS loop will not exceed 2300 psig. The results are consistent with and continue to comply with the current HNP licensing basis/acceptance requirements. The results obtained with the Δ75 replacement steam generators at the uprated power of 2912.4 MWt are applicable to operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt at corresponding values of T_{avg} and steam generator tube plugging level.

The revised RCS pressures based on 2912.4 MWt are applicable to operation of HNP at the current NSSS power rating of 2787.4 MWt since the RCS pressure analysis was not based on NSSS power but was dependent upon the RCS temperatures and flow rates.

4.1.1.3 RCS Volume, Mass, and Boron Inventory Analysis

4.1.1.3.1 Introduction

Westinghouse has evaluated the RCS volume, mass, and boron inventory changes that result from the SGR/Uprating plant configuration and conditions. The evaluation considered typical conditions that may occur in the RCS from operating at full power to shutdown mode and no Residual Heat Removal (RHR) trains in service to start-up conditions. The purpose of this calculation was to determine the volume of reactor coolant required for each case. Once the volume had been calculated, the mass and boron inventory of the reactor coolant was determined.

4.1.1.3.2 Description of Analyses and Evaluations

For the evaluation, the relevant systems that impact the volume were determined, and the volume of each applicable system was then calculated. For Mode 1 and Mode 2 operations, a 3-percent thermal expansion was included in the calculation of the RCS volume.

The RCS active volume for each case was the sum of the volumes for each applicable system. The mass was calculated as the product of the volume and the density of the reactor coolant at the temperature of the system.

At full power, the temperature varies throughout the RCS; therefore, the mass was determined for each of the temperature regions in the RCS.

For other RCS configurations and conditions, the RCS temperature was assumed to be relatively uniform. Therefore, the average RCS temperature was used to determine the coolant mass in the RCS. The coolant mass in the CVCS and RHRS were calculated. The total coolant mass was then calculated from the sum of the masses in each system.

The boron mass was then calculated.

4.1.1.3.3 Acceptance Criteria

There are no acceptance criteria for the results of the evaluation. The results are used as input to other efforts where additional evaluations or analyses are performed.

4.1.1.3.4 Results and Conclusions

This evaluation demonstrates the relationship between the RCS volume and boron mass in the system. These RCS volumes will be used in the boron dilution analysis, LTOP analysis, and operating procedure guidance.

This analysis was evaluated at the same temperatures for vessel outlet, core average, and vessel/core outlet as in the original basis for the current NSSS power of 2787.4 MWt. Therefore, the results obtained with the Δ75 replacement steam generators at the uprated power of 2912.4 MWt bound operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

4.1.1.4 RCS Heat Capacity

4.1.1.4.1 Introduction

A re-evaluation of the RCS heat capacity was required to reflect the plant configuration of HNP for the SGR/Uprating program. The parameters for this calculation included the masses and specific heat values of the primary side (reactor coolant, the nuclear fuel material, the vessels and piping) and the secondary side of the steam generator (feedwater, steam and piping). This result was incorporated into the revised RHR cooldown analysis (subsection 4.1.4) that was performed for the SGR/Uprating program.

4.1.1.4.2 Description of Analyses and Evaluations

The masses of the reactor coolant, feedwater and steam in the steam generator were determined by their volumes at 350°F. The mass of each parameter was multiplied by its respective specific heat. The heat capacities of the parameters were added to obtain the RCS total heat capacity.

4.1.1.4.3 Acceptance Criteria

There are no acceptance criteria for the results of the evaluation. The results are used as input to other efforts where additional evaluations or analyses are performed.

4.1.1.4.4 Results and Conclusions

For calculating the RCS total heat capacity, each mass was multiplied by its respective specific heat. The individual heat capacities were added to obtain a total RCS heat capacity of 1.29 MBtu/ °F.

The results are consistent with and continue to comply with the current HNP licensing basis/acceptance requirements. The results obtained with the Δ75 replacement steam generators at the uprated power of 2912.4 MWt bound operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

The revised RCS heat capacity based on 2912.4 MWt is bounding for operation of HNP at the current NSSS power rating of 2787.4 MWt since the RCS heat capacity analysis was not based on NSSS power but was dependent upon the mass and heat capacity of each component.

4.1.1.5 Pressurizer Spray System

4.1.1.5.1 Introduction

The purpose of this evaluation is to determine the effect of the SGR/Uprating conditions on the capability of the pressurizer spray system in HNP. The pressurizer spray flow rate is affected by changes in the reactor coolant best estimate flow rate, which depends on the steam generator tube plugging level and the associated effect on the pressure drop across the reactor vessel. The effect on the hydraulics of the RCS of the Siemens fuel design currently used in HNP was also considered in this analysis.

4.1.1.5.2 Description of Analyses and Evaluations

A model of the HNP pressurizer spray system was developed using a Bernoulli balance method. This model was used to analyze the hydraulic performance of the spray system for the RCS conditions associated with the SGR/Uprating.

The driving head for the spray flow is the differential pressure between the surge line connection in the hot leg and the spray line connection in the cold leg. The velocity head of the loop flow and the static head in the pressurizer add to the spray driving force. Once the head loss components have been determined, the pressurizer spray flow can be calculated.

The calculated pressurizer spray flow rate for the hydraulic conditions associated with the steam generator replacement and power uprating can then be compared to the minimum required flow rates to determine the acceptability of the system performance.

4.1.1.5.3 Acceptance Criteria

The design basis functional requirement for the pressurizer spray system is to prevent challenges to the pressurizer PORVs for a 10-percent step change in load. The minimum design basis flow rate of the HNP pressurizer spray system with both pressurizer spray flow control valves full open was 700 gpm. Although this analysis has demonstrated that flow rates in excess of 700 gpm can pass through the pressurizer spray line with the spray flow control valves full open, flow rates greater than 700 gpm are not required and are not expected to occur. The spray flow control valves will modulate open upon demand from the pressurizer pressure controller until the increase in pressurizer pressure due to the load decrease has been arrested. Therefore, the design flow rate of 700 gpm is the appropriate value for the transient structural analysis of the pressurizer spray nozzle.

4.1.1.5.4 Results and Conclusions

This analysis has determined that the design pressurizer spray flow rate of 700 gpm can be obtained in HNP for the conditions associated with the steam generator replacement and power uprating for the range of 0 to 10-percent steam generator tube plugging.

The results for the SGR/Uprating are consistent with and continue to comply with the current HNP licensing basis/acceptance requirements. The same RCS best estimate flow rate and core pressure drop apply to the steam generator replacement at the current power level. Therefore, the results obtained with the Δ75 replacement steam generators at the uprated power of 2912.4 MWt bound operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

4.1.1.6 Pressurizer Relief Tank Level Alarm Setpoints

4.1.1.6.1 Introduction

The SGR/Uprating program for HNP has prompted a reevaluation of the RCS. The purpose of this analysis is to review the pressurizer relief tank (PRT) level alarm setpoints. The original calculation determined that the alarms should be positioned at 83 percent for the high alarm and 64 percent for the low alarm of the 100-in. range. The objective of this analysis is to determine the revised level alarm setpoints for the PRT under the SGR/Uprating conditions.

4.1.1.6.2 Description of Analyses and Evaluations

The steam discharge calculation assumed that the PRT was capable of quenching 110 percent of the full-load pressurizer steam volume. A steam volume of 577 ft^3 included a 3-percent thermal expansion. The mass of steam in the pressurizer can be determined by dividing the volume by the specific volume.

The minimum volume was obtained by considering the enthalpy balance with conservatism. The water volume was converted from cubic feet to gallons. A conversion that relates water volume to liquid level for a horizontal tank was used to determine the water level in the PRT. Seven inches were subtracted from the water level to adjust for the height of the tank below the tap. For HNP, 3-percent trip accuracy was incorporated into the calculation. For the low-level setpoint, the accuracy was added; however, for the high-level alarm it was subtracted.

4.1.1.6.3 Acceptance Criteria

The low-level alarm setpoint in the PRT was determined such that the water temperature would not exceed the limit of 200°F after steam discharged.

PRT pressure can not exceed 50 psig after the design basis discharge with the selected high-level alarm setpoint.

4.1.1.6.4 Results and Conclusions

The result of the analysis yielded that the level alarm instruments should be positioned at 83 percent and 62 percent for high and low, respectively, of the PRT.

The results are consistent with and continue to comply with the current HNP licensing basis/acceptance requirements. Therefore, the results obtained with the Δ75 replacement steam generators at the uprated power of 2912.4 MWt bound operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

The revised PRT level alarm setpoints based on 2912.4 MWt are bounding for operation of HNP at the current NSSS power rating of 2787.4 MWt, as the PRT level alarm setpoints were not based on NSSS power, but were dependent upon the initial full-power pressurizer steam volume and operating conditions of the PRT, which are the same for both power ratings.

4.1.2 Chemical and Volume Control System

The Chemical and Volume Control System (CVCS) is a key auxiliary support system to the RCS. The CVCS is physically connected to the RCS and is normally in operation. Its primary design function is to maintain RCS water inventory, boron concentration, and water chemistry. Other RCS support functions include purification and seal injection flow to the reactor coolant pumps (RCPs). Key functions of the CVCS are as follows:

- RCS Pressure Boundary Integrity — Portions of the CVCS that comprise the reactor coolant pressure boundary are designed for expected RCS operating conditions to minimize the probability of pressure boundary failure.

- Boration — Consistent with the requirements of 10 CFR 50 Appendix A (General Design Criteria), the CVCS provides means for reactivity control independent of control rods. This function is accomplished via delivery of boric acid solution, which is a neutron absorber, into the RCS.

- Containment Isolation — Following design basis events, CVCS connections to the RCS that penetrate containment, with the exception of the RCP seal injection line, are automatically isolated during all events that require containment isolation.

The revised RCS operating conditions at the SGR/Uprating conditions are provided in Section 2 of this report. The primary RCS parameter changes include revised steam generator parameters, an increase in core power, allowable operating range for RCS T_{avg}, and reduced Thermal Design Flow.

The effects of the revised operating conditions on CVCS functions are described within this section.

4.1.2.1 Chemical and Volume Control System/ Boron Thermal Regeneration System Heat Exchanger Performance

4.1.2.1.1 Introduction

The Westinghouse CVCS design files for HNP were reviewed to identify existing analysis and inherent assumptions. An evaluation of the performance of the CVCS and Boron Thermal Regeneration System (BTRS) heat exchangers at the SGR/Uprating plant conditions is summarized below. Other existing CVCS performance analyses that remain applicable to the SGR/Uprating conditions are described in subsequent sections of 4.1.2.

4.1.2.1.2 Description of Analyses and Evaluations

The effect of the revised RCS operating temperatures on the following CVCS/BTRS heat exchangers was analyzed:

- Regenerative Heat Exchanger
- Letdown Heat Exchanger
- Excess Letdown Heat Exchanger
- Seal Water Heat Exchanger
- Letdown Reheat Heat Exchanger

The performance of the moderating heat exchanger and the letdown chiller heat exchanger was not evaluated in this analysis for the following reason. The BTRS contains the moderating, letdown chiller, and letdown reheat heat exchangers. The letdown stream, which flows through the BTRS demineralizers, is extracted from, and returns to, the letdown line at points downstream of the letdown heat exchanger. Therefore, as long as the letdown heat exchanger is capable of reducing the letdown temperature to 115°F or lower, there will be no change in the terminal temperatures or heat loads for the BTRS heat exchangers. Additionally, the BTRS heat exchangers do not utilize component cooling water because the moderating and letdown reheat heat exchangers are regenerative types of heat exchangers that recover heat from, and reject heat to, other parts of the CVCS/BTRS systems that are at higher or lower temperature. The letdown chiller heat exchanger utilizes a closed-loop chilled water system for cooling water. The only heat exchanger that is potentially affected by the SGR/Uprating is the letdown reheat heat exchanger.

With respect to CCWS operating temperatures for those CVCS/BTRS heat exchangers cooled by CCW, the maximum expected CCWS supply temperature provided to CVCS components was compared to allowable operating limits (see Section 4.1.6). These operating limits were identified or confirmed in the analysis of the performance of the RHRS during plant cooldown (see Section 4.1.4) and in the thermal analysis of the CCWS (see Section 4.1.6).

4.1.2.1.3 Acceptance Criteria

In the assessment of CVCS operation at revised RCS operating temperatures, the maximum expected RCS T_{cold} must be less than or equal to the applicable CVCS design temperature and less than or equal to the heat exchanger design inlet operating temperature. The former criterion supports the functional operability of the system and its components. The latter criterion provides for heat exchanger design operating conditions that remain bounding.

The letdown line fluid temperature shall be reduced to a temperature less than or equal to 110°F downstream of the letdown heat exchanger and the seal water heat exchanger. The tube-side

outlet temperature of the excess letdown heat exchanger was originally designed to be less than or equal to the original design tube-side outlet temperature of 165°F. The current HNP procedures may control the excess letdown such that the tube side outlet temperature is 174°F. Both limits were considered in the evaluation.

In the assessment of CVCS operation at revised CCWS operating temperatures, the maximum expected CCWS supply temperature must be less than or equal to allowable operating limits, which are 105°F for normal operation and 125°F during plant cooldown. This criterion supports the overall functional operability of the system and the components serviced by the CCWS.

The temperature of the component cooling water leaving the CVCS heat exchangers that are cooled by the CCWS shall remain below the levels at which local cavitation or voiding could occur in the component cooling water system. It was assumed that 50 degrees of subcooling provides adequate margin of local subcooling.

The CVCS heat exchanger terminal temperatures shall remain within design temperature limits specified for the tube and shell sides of the units. The differences in terminal temperatures for the heat exchangers shall be acceptable from the aspect of component stress limits for normal operation.

Heat exchanger flowrates shall remain within mechanical design constraints of the heat exchanger unit with respect to pressure differential and vibration.

4.1.2.1.4 Results and Conclusions

The results of the CVCS heat exchanger performance analysis support the conclusion that the operating flow rates and temperatures are very close to current values, and are bounded by the heat exchanger design conditions.

The operation of the CVCS heat exchangers at the revised RCS temperature conditions was reviewed and the results presented in the previous subsection. The conclusion from this assessment is that the design of the CVCS heat exchangers will remain adequate for the conditions associated with the SGR/Uprating. This portion of the CVCS system supports operation at the current power with steam generator replacement as well as steam generator replacement in conjunction with power uprating. Other aspects of the design of the CVCS are addressed in the remainder of Section 4.1.2.

4.1.2.2 Letdown Line Performance

4.1.2.2.1 Introduction

Westinghouse performed an analysis of the letdown line performance to: (1) verify the adequacy of valve PCV-145 to provide the maximum letdown flowrate of 120 gpm, (2) verify that the

setpoints of relief valves 1-8117 and 1-8119 are not exceeded during normal operation, and (3) determine the nominal setpoint for pressure transmitter PT-145.

4.1.2.2.2 Description of Analyses and Evaluations

The following analyses were performed: (1) verification of adequacy of PCV-145 control range, (2) protection of relief valve set pressures, and (3) determination of nominal setpoint for PT-145.

4.1.2.2.3 Acceptance Criteria

Acceptance criteria applicable to this analysis include: (1) the calculated C_v for valve PCV-145 must be within the range of 2 – 40, (2) the normal operating pressure at the inlet of relief valve 1-8117 must be less than 600 psig, and (3) the normal operating pressure at the inlet of relief valve 1-8119 must be less than 300 psig.

4.1.2.2.4 Results

The results of the analysis are:

1. The maximum required C_v of valve PCV-145 is 9.3.
2. The minimum required C_v of valve PCV-145 is 6.1.
3. The normal operating pressure at the inlet of relief valve 1-8117 is 456.8 psig.
4. The normal operating pressure at the inlet of relief valve 1-8119 is 242.4 psig.
5. The nominal setpoint for PT-145 is 400 psig.

4.1.2.2.5 Conclusions

The performance of the letdown line in the HNP CVCS design, as determined in the above analysis, is projected to meet all the applicable acceptance criteria and is acceptable. There are no changes associated with the implementation of the SGR/Uprating program that would invalidate this analysis or its conclusions.

This portion of the CVCS system supports operation with the replacement steam generators at the current power level as well as at the uprated power level.

4.1.2.3 Charging System Evaluation

4.1.2.3.1 Introduction

Westinghouse performed an analysis of the charging system performance to: (1) determine the adequacy of the charging flow control valve FCV-122 C_v range, (2) determine the net positive

suction head (NPSH) available to the centrifugal charging pump from the VCT, and (3) determine the RCS dilution flow rates at power and during refueling.

4.1.2.3.2 Description of Analyses and Evaluations

The following analyses were performed: (1) verification of adequacy of FCV-122 maximum C_v required, (2) verification of adequacy of available NPSH to centrifugal charging pump, and (3) determination of dilution flow rates to the RCS at power and during refueling.

4.1.2.3.3 Acceptance Criteria

Acceptance criteria applicable to this analysis include: (1) the calculated C_v for valve FCV-122 must be within the range of 0.2 – 22, (2) the NPSH available to the centrifugal charging pump must exceed the required NPSH at the maximum pump flow rate (12 ft at 189 gpm), and (3) the acceptance criteria for the RCS dilution flow rates are determined separately in the inadvertent boron dilution analysis (See Section 6.2).

4.1.2.3.4 Results

The results of the analysis are:

1. The maximum required C_v of valve FCV-122 is 7.21.

2. The NPSH available to the centrifugal charging pump from the VCT at a flow rate of 189 gpm is 50.23 ft.

3. The RCS dilution flow rate for normal power operation is 216 gpm.

4. The RCS dilution flow rate during refueling is 590 gpm.

4.1.2.3.5 Conclusions

The performance of the charging line in the HNP CVCS design, as determined in the above analysis, meets all the applicable acceptance criteria and is acceptable from the system hydraulic standpoint.

There are no changes associated with the implementation of the SGR/Uprating program that would invalidate the analysis or its conclusions.

The basis for the flowrates assumed in the RCS boron dilution analysis for the SGR/Uprating program is consistent with the design basis of the CVCS charging and makeup systems.

This portion of the CVCS system supports operation of the replacement steam generators at the current power level as well as at the uprated power level.

4.1.2.4 Excess Letdown Flow Performance

4.1.2.4.1 Introduction

The purpose of this analysis is to determine the adequacy of valve HCV-137 in the excess letdown line to provide the design flow rate of 25 gpm with the excess letdown flow directed to the VCT.

4.1.2.4.2 Description of Analyses and Evaluations

The pressure drop through the excess letdown line at the design excess letdown flow rate (25 gpm) was calculated. Using the calculated pressure drop information, the minimum pressure drop across valve HCV-137 was calculated. The maximum required C_v value for valve HCV-137 was then determined and compared to the maximum available C_v of the valve to determine that the valve will pass the design excess letdown flow rate.

4.1.2.4.3 Acceptance Criteria

The calculated C_v for valve HCV-137 at a flow rate of 25 gpm must be less than the maximum value of 1.5.

4.1.2.4.4 Results

The results of this analysis indicate that a C_v value of 0.55 is required for valve HCV-137, which is less than the maximum C_v value of 1.5.

4.1.2.4.5 Conclusions

The performance of the excess letdown line in the HNP CVCS design meets all the applicable acceptance criteria and is acceptable. There are no changes associated with the implementation of the SGR/Uprating program that would invalidate this analysis or its conclusions.

This portion of the CVCS system supports operation with the replacement steam generators at the current power level as well as at the uprated power level.

4.1.2.5 Boric Acid System Performance

4.1.2.5.1 Introduction

Westinghouse performed an analysis of the boric acid system performance to: (1) determine the boric acid flow capacity for normal makeup into the VCT or to the VCT outlet (bypassing the tank), (2) determine the ability of the boric acid system to provide emergency boration flow to the suction of the centrifugal charging pump and the NPSH that the boric acid transfer pump (BATP) provides to the centrifugal charging pump in this mode of operation, and (3) determine the maximum runout flow rate of the BATP in the emergency boration mode.

4.1.2.5.2 Description of Analyses and Evaluations

The following analyses were performed: (1) verification of adequacy of FCV-113A maximum C_v required for normal makeup, (2) verification of emergency boration flow rate and NPSH provided to centrifugal charging pump, and (3) determination of maximum boric acid transfer pump runout flow rate in the emergency boration mode of operation.

4.1.2.5.3 Acceptance Criteria

Acceptance criteria applicable to this analysis include: (1) for normal makeup, the calculated maximum C_v for valve FCV-113A must be less than 22.8, (2) the NPSH available to the centrifugal charging pump must exceed the required NPSH at the assumed emergency boration flow rate (10 ft at 61 gpm), and (3) the runout flow rate of the boric acid transfer pump in the emergency boration mode of operation must be less than 150 gpm.

4.1.2.5.4 Results

The results of the analysis are: (1) the maximum required C_v of valve FCV-113A is 11.6 for normal makeup to the VCT operating at a pressure of 30 psig, (2) the NPSH available to the centrifugal charging pump from the BATP with the VCT isolated at a flow rate of 61 gpm in the emergency boration mode of operation is 167.7 ft, and (3) the resistance of the emergency boration flow path limits the maximum flow rate of the boric acid transfer pump to less than 135 gpm.

4.1.2.5.5 Conclusions

The performance of the boric acid system in the HNP CVCS design, as determined in the above analysis, meets all the applicable original acceptance criteria and is acceptable from the system hydraulic standpoint.

There are no changes associated with the implementation of the SGR/Uprating program that would invalidate this analysis or its conclusions.

It has also been confirmed that the refueling concentration for the core design at the uprated power level does not exceed 2600 ppm boron.

This portion of the CVCS system supports operation with the replacement steam generators at the current power level as well as the uprated power level.

4.1.3 Safety Injection System

4.1.3.1 Introduction

The Safety Injection System (SIS) is an Engineered Safeguards System used to mitigate the effects of postulated design basis events. The basic functions of this system include providing short-term and long-term core cooling, and maintaining core shutdown reactivity margin. The SIS is also referred to as the Emergency Core Cooling System (ECCS).

At the Harris Nuclear Plant (HNP), the SIS is comprised of three subsystems. The passive portion of the system is comprised of three accumulator vessels, which are connected to each of the Reactor Coolant System (RCS) cold leg pipes. Each accumulator contains borated water under pressure (nitrogen cover gas). The borated water automatically injects into the RCS when RCS pressure drops below the operating pressure of the accumulators.

The active portion of the SIS is comprised of a high-pressure injection subsystem and a low-pressure injection subsystem. Both subsystems utilize centrifugal pumps, which automatically start following generation of a Safety Injection (SI) signal. There are three High Head Safety Injection (HHSI) pumps, of which two are normally available for safety injection. The third pump is an installed spare to which the power supply is removed. The HHSI pumps provide the normal charging function and are normally aligned to take suction from the volume control tank. Upon receipt of an SI signal, the pump suction is aligned to the Refueling Water Storage Tank (RWST) and isolated from the volume control tank. The HHSI pump discharge is simultaneously aligned from the charging header to the cold-leg safety injection system header. Each operating charging pump has a miniflow line, which recirculates fluid from the pump discharge back to the pump suction. The normal miniflow path is automatically isolated during the initial phases of safety injection and the HHSI pump is either realigned to an alternate miniflow path or pump miniflow is terminated, depending on RCS pressure. In either case, the HHSI pumps continue to provide injection flow to the reactor coolant pump number one seals during post-accident operations. The alternate miniflow path ensures that the HHSI pumps are not "deadheaded" if RCS pressure increases above the pump shutoff head during the injection phase of ECCS operation.

The Residual Heat Removal System (RHRS) is used as the low-pressure subsystem. This Low Head Safety Injection (LHSI) subsystem is initially aligned to take suction from the RWST and deliver borated water to each of the RCS cold legs. This system has two parallel low-head pumps available for duty. LHSI injection flow would occur as soon as RCS pressure drops below the RHRS "Cut-In" pressure (RHRS pump shutoff head adjusted for suction pressure provided by the static head of water in the RWST).

As the design basis event proceeds, the RWST borated water inventory decreases as water is transferred to the RCS and/or containment building. Upon depletion of a majority of the RWST inventory, the operating HHSI and LHSI pumps are required to be realigned, using the sump as a

water source, to support the cold-leg recirculation mode of operation. Upon an RWST low-low level condition, the suction of the RHRS pumps is manually realigned to the containment sump. The sump isolation valves open automatically on a low-low level condition. Component cooling water flow is supplied to the RHR heat exchanger to cool the sump fluid. During the recirculation phase of the LOCA accident, the RHR pumps continue to provide flow to the RCS cold legs. In addition, the suction of the HHSI pumps is manually realigned to take suction from the RHR pump discharge. Long-term core cooling is provided by the LHSI system (RHR) heat exchangers.

After a pre-determined time to prevent boron precipitation in the core, the SIS is realigned to support the hot-leg recirculation mode of operation. Subsequent recirculation operation requires aligning the recirculated coolant flow alternately to the cold legs and the hot legs of the RCS.

In an immediate response to a design basis event, the SIS is designed to perform its safety functions, assuming a loss-of-offsite power, and considering a single active failure. An example of a postulated single failure is the failure of an emergency diesel generator to start. This failure would preclude a train of SIS pumps from operating. In the long-term (beyond 24 hours), if an active failure has not occurred, a system leak (i.e., passive failure specifically identified in the licensing bases) could occur which may require operator actions to realign the system to support adequate performance. Several alternate flow paths exist so that long-term core cooling can be provided should a passive failure occur.

4.1.3.2 Description of Analyses and Evaluations

A review of the Westinghouse Precautions, Limitations and Setpoints (PLS) Document for HNP, relevant to the SIS, was performed and no changes were identified to support implementation of the SGR/Uprating conditions.

The latest Westinghouse calculations of record for the SIS performance flows were provided for use in the NSSS Accident Analysis in Section 6.0. Refer to the applicable summaries of the safety analyses in Section 6.0 for a description of the results applicable to those analyses.

Additional SIS injection flow data provided for the SGR/Uprating Project were used in the following analyses: Steam Generator Tube Rupture, large break LOCA mass and energy releases, and main steamline break mass and energy releases (inside and outside of containment).

Minimum RHR and HHSI flow rates (one RHR pump and one HHSI pump) for the cold leg recirculation mode of ECCS operation were calculated. Hot leg and cold leg recirculation flow rates were used in the long-term core cooling analysis for the HNP SGR/Uprating Project (see Section 6.1.3).

4.1.3.3 Acceptance Criteria

The general acceptability of the SIS operating parameters defined for the SGR/Uprating project are documented in the individual plant safety analyses that utilize such inputs. Refer to Section 6.0.

4.1.3.4 Results and Conclusions

In general, the specified changes in RCS operating conditions addressed by the SGR/Uprating project (higher core power and range in hot full-power T_{avg}) as outlined in Section 2 of this report have no direct effect on the overall performance capability of the SIS. These systems are capable of delivering an acceptable range of calculated flow performance (minimum and maximum) as determined by interfacing system/structure operating conditions (RCS pressure, containment pressure, etc.).

The acceptability of the existing and newly calculated SIS operating parameters is documented in the various individual plant safety analysis results as summarized in Section 6 of this report. The SIS supports operation at the current power and steam generator replacement as well as steam generator replacement in conjunction with power uprate.

4.1.4 Residual Heat Removal System

4.1.4.1 RHR Cooldown Analysis

4.1.4.1.1 Introduction

The Residual Heat Removal System (RHRS) is a dual-function system. During normal power operation, the system is in a stand-by mode to support its Engineered Safeguards function (i.e., safety injection). During the second phase of plant cooldown and the plant shutdown mode of operation, the RHRS is used to remove RCS sensible and core decay heat. The auxiliary feedwater and main steam systems are used for the RCS heat removal during the first phase of plant cooldown and may supplement the second phase of plant cooldown. This section discusses the RHRS normal functions (i.e., heat removal). The Engineered Safeguards functions of the RHRS are discussed in Section 4.1.3, Safety Injection System.

The RHRS is comprised of two centrifugal pumps, two heat exchangers, interconnecting piping and instrumentation. With the RHRS in operation, each RHR pump takes suction from an RCS hot leg and recirculates the flow back to each of the RCS cold legs. System flow passes through the tube side of the RHR heat exchangers (shell and tube design). Cooling flow to the RHR heat exchangers (shell side) is provided via the Component Cooling Water System (CCWS), which in turn, is cooled by the Service Water (SW) System. The CCWS is comprised of three pumps and three heat exchangers. Normally, two CCW pumps and two CCW heat exchangers are in service during cooldown.

In addition, the RHRS is equipped with two relief valves for protection against overpressurization. (Note that Section 4.3.1.5 discusses the LTOP analysis and the use of the RHR relief valves to protect the RCS against overpressure events at low temperatures.)

The maximum heat removal demand on the RHRS occurs during the plant cooldown mode of operation when RCS sensible heat (e.g., metal mass), core decay heat, and heat input from a Reactor Coolant Pump (RCP) must all be removed to support RCS cooldown. In addition, operating restrictions are imposed on the maximum allowable CCWS temperature and flow during cooldown, which can also restrict RHRS heat removal capability.

The overall RHRS heat removal capability can vary significantly depending on system equipment availability, cooling support system equipment availability, cooling support system flows, and SW system inlet temperature. In general, RHRS thermal heat removal capability becomes more restricted when operating conditions change as outlined below.

- Higher RCS heat loads
- Lower RHRS flows
- Lower CCWS flow to the RHRS heat exchanger

- Lower SW flow to the CCWS heat exchanger
- Higher CCWS auxiliary heat loads

4.1.4.1.2 Description of Analyses and Evaluations

Westinghouse evaluated the cooldown analysis. The flow rates were determined as described in BOP Licensing Report Section 2.6. The evaluation used a maximum flow rate of 12,500 gpm of CCW in the CCW heat exchanger. See Section 4.1.6 for additional information concerning the thermal performance of the CCWS.

The Spent Fuel Pool (SFP) heat exchanger heat load also affects the capability of the RHRS to cool the system. A SFP heat load of 17.9 MBtu/hr was evaluated. The higher core power levels and the additional SFP heat load will increase the RHR heat loads, which must be removed during plant cooldown and shutdown conditions. Cooldown was modeled with a maximum allowable supply CCW temperature of 120°F. In addition, the design UA for the RHR heat exchanger and the CCW heat exchanger was adjusted by 0.5 percent to account for future tube plugging.

RHRS thermal performance was calculated for each of the following three cooldown scenarios:

1. The ability of the RHRS to cool down the RCS with all equipment operating to a refueling condition (140°F). Note that RHRS operation with all equipment available (including support systems) is referred to as a "normal" plant cooldown within the context of this section.

2. The ability of the RHRS to cool down the RCS under limiting equipment availability to a cold shutdown condition (200°F). Note that RHRS operation with one subsystem of equipment available (including support systems) is referred to as a "single train" plant cooldown within the context of this section.

3. Single train cooldown without RCP heat load, referred to as "natural circulation" within the context of this section.

Selected cases were run at both the current licensed rated thermal power of 2775 MWt and the uprated thermal power of 2900 MWt. For scenario 1, the RHRS major components were originally sized to achieve a targeted (desired) overall cooldown from system cut-in (which occurs 4 hours after reactor shutdown) to a refueling RCS temperature (140°F) in approximately 16 hours (20 hours total duration). This normal cooldown (with all cooling equipment available) was reanalyzed at the higher core power level.

For scenario 2, the Standard Review Plan (SRP) requires that the RHRS be capable of cooling the RCS from Hot Standby (350°F) to 200°F within a reasonable time period in the event that one CCWS train is unavailable.

Following are the key inputs used in the various thermal (cooldown) analyses:

• RHRS Cut-in Temperature	- 350°F
• CCWS Maximum Allowable Temperature	- 120°F for normal and single train cooldown
• SW System Maximum Inlet Temperature	- 95°F during normal and single train cooldown
• RCP Heat Input Term to the RCS/RHRS	- 54.96 x 10^6 Btu/hr (for three RCPs between 350°F and 200°F)
	- 19.67 x 10^6 Btu/hr (for one RCP below 200°F)
	- 18.05 x 10^6 Btu/hr (avg. for one RCP between 350°F and 200°F)
• RCS Heat Capacity	- 1.29 E+6 Btu/°F
• SW System Flow to the CCWS Heat Exchangers	- 10,000 gpm to each heat exchanger

For the previously mentioned cooldown scenarios, various cases were analyzed. Provided below is a listing of the specific cases analyzed for each cooldown scenario.

- Normal Cooldown - All equipment available; three RCPs in service from 350°F to 200°F and one RCP in service from 200°F to 160°F. (Two RHRS pumps/heat exchangers and two CCWS pumps/heat exchangers were assumed during this scenario.)

- Single Train Cooldown (cooling equipment out of service) - One train unavailable (one RHRS pump/heat exchanger and one CCWS pump/heat exchanger and one RCP in operation)

- Natural Circulation (with no RCP heat load) - One train unavailable and RCP out of service due to a loss-of-offsite or onsite power (one RHRS pump/heat exchanger and one CCWS pump/heat exchanger)

4.1.4.1.3 Acceptance Criteria

When all cooling equipment is available, the RHRS is expected to perform a normal cooldown. The original RHRS equipment sizing criteria (20 hour cooldown from Hot Standby no-load

conditions at the time of plant shutdown to 140°F, assuming 4 hours for cooldown using the main steam system to Hot Shutdown conditions (RHRS initiation) at 350°F and 16 hours of cooldown using the RHRS to 140°F) was selected based partly on economic considerations. As such, there is not a specific design basis acceptance criterion for the normal cooldown.

4.1.4.1.4 Results and Conclusions

The following are the results and conclusions drawn for the analyzed scenarios:

1. RHRS Cut-in

   It was determined that the RHRS is capable of accepting the RCS cooldown heat loads at the uprated conditions. (See Tables 4.1.4.1-1 and 4.1.4.1-2.) However during the single train cooldown, it was determined that heatup of the RCS above 350°F may occur upon initiation at 6 hours after shutdown, since an RCP remains in operation. The steam generator PORVs can sustain cooling until the RHRS is capable of maintaining the RCS temperature under 350°F. The time at which a single train of RHRS can accept the decay heat and RCP heat load is identified in Table 4.1.4.1-2.

2. Normal Cooldown

   With all cooling equipment trains available and assuming RHRS operation is initiated at 4 hours after plant shutdown, the maximum calculated duration to cool down the RCS from 350° to 200°F (Cold Shutdown) is less than 6 hours. The total calculated duration (8.7 hours) is well within the 30-hour Technical Specification cooldown time.

   Cooldown to temperatures below 200°F is an economic consideration and is not governed by the Technical Specifications. With all cooling equipment trains available, the maximum calculated time to cool down the RCS from 200° to 140°F (Refueling) is less than 20 hours. Any extension of the cooldown time beyond the original design basis is primarily an economic factor. This analysis demonstrates that RHRS is capable of cooling the RCS within 24 hours after shutdown. Results for the normal cooldown analysis are shown in Table 4.1.4.1-1.

3. Single Train Cooldown (Equipment Unavailable)

   If plant cooldown is initiated with only one train of equipment available, and assuming RHRS operation is initiated at 6 hours after plant shutdown, the RHRS can cool the RCS from 350°F to 200°F within the times specified in Table 4.1.4.1-2. However, it was determined for both the SGR and SGR/Uprating system configurations that the RCS temperature could not be maintained below 350°F immediately after RHR cut-in if RHR operation was begun at 6 hours after shutdown. Therefore, cooling would have to be sustained via the SG PORVs until the RHRS was capable of maintaining the RCS

temperature to less than 350°F. It has been determined that RHRS alone cannot remove the decay heat and RCP heat until the times listed in Table 4.1.4.1-2.

4. Natural Circulation (RCP Not in Service)

In the event the plant experiences a loss-of-offsite power, the system must be capable of bringing the reactor to cold shutdown. Upon initiation at 17 hours after shutdown, the RHRS can cool the RCS from 350°F to 200°F within 10 hours with only one train of equipment available. The use of 6 hours is very conservative, since the natural circulation cooldown analysis concludes that when natural circulation and only safety-related equipment are relied upon for the cooldown from 557°F to 350°F, RHR cut-in would occur no sooner than 17 hours from reactor shutdown.

Based on the above discussions, it can be concluded that the operation of the HNP RHRS at thermal uprate RCS operating conditions should not impact its ability to perform intended decay heat removal functions.

The analyzed scenarios showed that the cooldown times are sensitive to the increased core thermal power, increased SFP heat load, and the CCWS flow. During normal operation, actual cooldown times are expected to be shorter than those calculated in the analyzed scenarios, since actual CCWS performance and SW System temperatures are not expected to be at the conservative analytical limits assumed in the analyzed scenarios. In addition, the SFP decay heat load was also conservatively specified. (Refer to the BOP Licensing Report Section 2.9.)

The results are consistent with and continue to comply with the current HNP licensing basis/acceptance requirements. The results obtained with the Δ75 replacement steam generators at the uprated power of 2912.4 MWt bound operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt. Operation of HNP at the current NSSS power of 2787.4 MWt with the revised RHR parameters based on 2912.4 MWt are bounding, since the current NSSS power is approximately 96 percent of the uprated NSSS power.

Table 4.1.4.1-1 Normal Cooldown Analysis Input and Results (Normal Cooldown from 350°F to 140°F)	
	SGR/Uprating
Reactor Power (MWt)	2900
SFP Load (Mbtu/hr)	17.9*
Time Cooldown Initiated (hr)	4
Max CCW Temp. (°F)	120
SW Temp. (°F)	95
RCP Load (Mbtu/hr)	54.96/19.7
Cooldown Time (hr)	23.3

Table 4.1.4.1-2 Single Train Cooldown Analysis Input and Results (Single Train Cooldown from 350°F to 200°F)		
	SGR	**SGR/Uprating**
Reactor Power (MWt)	2775	2900
SFP Load (Mbtu/hr)	17.9	17.9
Time Cooldown Initiated (hr)	17	20
Max CCW Temp. (°F)	120	120
SW Temp. (°F)	95	95
RCP Load (Mbtu/hr)	18.05	18.05
Cooldown Time (hr)	40.3	44

*The selection of the SFP heat load is discussed in the BOP Licensing Report Section 2.9.

4.1.5 Boron Recycle System

This section provides a system review to evaluate the effects of the SGR/Uprating on the Boron Recycle System (BRS).

4.1.5.1 System Function

The function of the BRS is to collect deaerated, tritiated, borated radioactive effluent from the primary plant systems. The BRS decontaminates and processes the effluent, producing 4 wt. percent boric acid solution and reactor makeup water that is of sufficient quality for reuse as recycled makeup to the Reactor Coolant System (RCS) and spent fuel pools.

4.1.5.2 System Design Bases

The original design basis of the BRS is summarized as follows:

1. The BRS shall collect and process reactor coolant effluent via the Chemical and Volume Control System (CVCS) letdown line from the following sources:

    - Dilution of the RCS to compensate for core burnup
    - RCS dilution/boration operations associated with four hot shutdowns in one fuel cycle
    - RCS dilution/boration operations associated with three cold shutdowns in one fuel cycle
    - RCS dilution/boration operations associated with one refueling shutdown in one fuel cycle

2. The BRS shall collect and process reactor coolant effluents from the following additional sources:

    - Leakoff type drains from equipment inside containment via the Reactor Coolant Drain Tank
    - Volume Control Tank (VCT), charging pump suction and residual heat removal pump relief valve discharges
    - Boric Acid Tank maintenance draindown via the CVCS boric acid blender
    - Refueling transfer canal maintenance draindown via the spent fuel pool pumps
    - Gas decay tank drains
    - Boron injection tank valve flush water
    - Primary system valve leakoff and equipment drains

3. The BRS shall process the reactor coolant effluent that is collected by the system and produce 4 wt. percent boric acid solution and reactor makeup water that are of sufficient purity for recycling back to the RCS. This design criterion assumes fission product concentrations in the reactor coolant effluent that are based on operation with cladding defects in 1 percent of the fuel assemblies.

4.1.5.3 Evaluation of SGR/Uprating Effects

The effect of the steam generator replacement on the RCS volume and mass and, consequently, on the sizing basis of the Recycle Holdup Tank (RHT) was evaluated. The original sizing criterion of the RHT is based on a design surge cold shutdown and startup at an RCS boron concentration of 200 ppm. The original sizing of the RHT takes credit for the operation of the BTRS in the RCS boration and dilution operations that are associated with the shutdown and startup to reduce the volume of reactor coolant effluent that is generated by these operations. The size of the existing RHT is evaluated using the following revised design basis for the SGR/Uprating conditions:

- RCS volume and mass are based on the installation of the Model Δ75 steam generators.

- The HNP BTRS demineralizers are used for deboration of the RCS to compensate for core burnup below a boron concentration of approximately 300 ppm. However, the BTRS is not used to dilute the RCS during plant startup after a cold shutdown, and no credit will be taken for use of the BTRS in this evaluation of the design basis RHT sizing.

- The BRS Recycle Evaporator is no longer used in HNP, but its functionality is provided by the Waste Evaporator, which has the same nominal capacity as the Recycle Evaporator.

4.1.5.4 Conclusion of System Review

This evaluation of the Boron Recycle System in HNP has described the system design bases and functional requirements. On the basis of this evaluation, it can be concluded that the design of the HNP BRS remains capable of performing its intended function for operation at the current power with steam generator replacement as well as steam generator replacement in conjunction with power uprate.

4.1.6 Component Cooling Water System

4.1.6.1 Introduction

The HNP SGR/Uprating requires the evaluation of the performance of the Component Cooling Water System (CCWS) for the new plant conditions. This evaluation involves confirmation that the CCWS can supply the required cooling water flow rates at the required cooling water temperatures for the heat loads on the system in each of several different plant operating situations. The loads on the system consist of those items of equipment in the various NSSS systems that require component cooling water. Westinghouse determined the heat load and cooling water requirements for the various items of equipment in those systems that were originally in the Westinghouse scope of supply. The component cooling water flow distribution depends on the hydraulic losses of the system, the system configuration, as well as the pump curve. (Refer to BOP Licensing Report Section 2.6.) Ten operating configurations were evaluated for which the CCWS will provide cooling. Based on the limiting system resistances (Shutdown at 350°F) and the modified pump performance, flow rates were calculated for the other operating configurations. For each configuration, the following flow scenarios were evaluated:

1. Minimum Flow - Based on increasing the capacity of pump beyond original rated capacity. The calculated flows were reduced by 6 percent to remove built-in margin for calculation and instrumentation uncertainty.

2. Maximum Flow - Based on increasing the capacity of CCW pump beyond original rated capacity. These flow rates were not adjusted.

This analysis determined the following information for the HNP SGR/Uprating:

- Component heat and flow load data for the CCWS
- CCW supply temperatures for 10 operating configurations
- Component outlet temperatures for thermal stress analysis

4.1.6.2 Description of Analyses and Evaluations

In general, there are two ways considered in this evaluation that the auxiliary heat exchangers can be impacted by the SGR/Uprating:

- Operation at higher power level generates a higher level of decay heat during shutdown or accident periods. This factor impacts the heat load for heat exchangers such as the RHR heat exchanger and the SFP heat exchanger, which are directly linked to decay heat generation rate, but it does not impact the other heat exchangers or items of equipment that are serviced by the CCWS.

- Replacement of the steam generators affects coolant volume and metal mass in the RCS. The change in RCS heat capacity affects the sensible heat released during plant cooldown. This "heat," along with the core decay heat and the RCP heat are transferred into the CCWS via the RHR heat exchanger.

A comparison of the uprated RCS conditions to original design conditions demonstrates that the changes in RCS operating temperatures are minor.

The method used for defining the heat loads imposed on the CCWS follows the general method used historically for the original design.

Operating Configurations of the CCWS

There are 10 defined operating configurations of the CCWS considered. These plant operating configurations are defined as follows:

1. Plant Startup at 350°F – This covers the heatup from cold shutdown (mode 5) or refueling (mode 6) to hot shutdown conditions (mode 4). At 350°F, the RHR system is in service.

2. Plant Startup at 557°F – This covers the period from hot shutdown (mode 4) to critical minimum load at operating plant pressure and temperature (mode 2). After the RHR system is isolated, the excess letdown and the letdown system are aligned for maximum flow rate to remove the coolant expansion that occurs during plant heatup.

3. Normal Operation – This includes power operation (mode 1) and operation at hot standby (mode 3) with the RCS at normal operating pressure and temperature.

4. Plant Shutdown at 350°F – This refers to the initiation of residual heat removal operations for plant cooldown at an RCS hot leg temperature of 350°F. The RHR system removes core generated decay heat, RCP work input into the RCS, and also the sensible heat released by reducing the temperature of the RCS primary side and secondary side metal and coolant.

5. Plant Shutdown at 140°F – This refers to the completion of plant cooldown to 140°F in the RCS. The RHR system is in service to remove decay heat and maintain stable RCS temperature.

6. Refueling – The RHR system is in service to remove core generated decay heat and maintain stable temperature in the RCS and in the refueling cavity for refueling operations. The refueling mode considers the plant situation prior to the time that spent fuel is moved from the reactor to the fuel pool.

7. Safety Injection – This corresponds to the injection configuration of safety injection operation with the safety injection pump suction aligned to the RWST. There is no heat

load imposed by the RHR heat exchangers in this configuration. The safety injection event is assumed to initiate a phase A containment isolation signal, which isolates CCW loads inside containment.

8. Recirculation at 243.5°F – During Recirculation at 243.5°F (peak sump temperature), it was assumed that all heat loads are isolated except for the RHR pump seal coolers and the RHR heat exchangers. The RHR heat load is conservatively estimated using an RHR flow rate of 4,500 gpm at 243.5°F through the tube side of the RHR heat exchanger.

9. Recirculation at 200°F - For Recirculation at 200°F, it was assumed that the fuel pool heat and flow loads are reinstated. The RHR heat load is conservatively estimated using an RHR flow rate of 4,500 gpm at 200°F through the tube side of the RHR heat exchanger.

10. Loss of Offsite Power - Loss-of-offsite power causes the reactor to trip. The major plant control systems return the RCS and steam generator secondary side to hot zero power condition (mode 3).

The aforementioned plant configurations are used where the heat loads and flows for the individual components are determined for the different plant configurations.

Components Serviced by the CCW system

The components serviced by the CCWS are listed in Table 4.1.6-1.

Calculated Performance of the CCW Heat Exchanger

For each combination of heat load and flow rate, an analysis was performed to determine the CCW heat exchanger shell inlet and return temperatures.

4.1.6.3 Acceptance Criteria

Acceptance criteria for this analysis are as follows:

1. The maximum actual CCW supply temperature shall be limited to 125°F when RCPs are operating during plant cooldown and to 105°F during normal plant operation. The original CCW supply temperature limit when the RCPs are operating during plant cooldown was 120°F.

2. The heat exchanger terminal temperatures shall remain within design temperature limits specified for the tube and shell sides of the units on the component data sheets.

3. The differences in terminal temperatures for the heat exchangers shall be acceptable from the aspect of component stress limits for normal operation.

4. The temperature of the component cooling water leaving the components cooled by the CCW system shall remain below the levels at which local cavitation or voiding could occur in the CCWS. Sufficient margin of subcooling shall be maintained. Engineering judgement indicates that 50 degrees of subcooling provides adequate margin of local subcooling.

5. Heat exchanger flow rates shall remain within mechanical design constraints of the heat exchanger unit with respect to pressure differential and vibration.

6. The CCWS shall be able to supply the required flow rates, and the flow coefficients for the modulation/throttle valves (either manual or automatic) shall be within the controllable ranges for the valve characteristics.

4.1.6.4 Results and Conclusions

Table 4.1.6-2 provides the CCW supply and component outlet temperatures for each configuration. The maximum allowable outlet temperature for the RCP upper bearing oil cooler is 198°F. Otherwise, the design temperature of the CCWS of 200°F is conservative.

All component outlet temperatures are within the above defined temperature limits for all plant operating configurations. The results of the component outlet temperature evaluation are acceptable.

The CCW supply temperatures exceed the original design basis of 105°F for Plant Start-up at 350°F, Plant Start-up at 557°F, Refueling, and 120°F for Recirculation at 243.5°F and Recirculation at 200°F. These results are acceptable because they can be justified as follows:

The Plant Start-up cases marginally exceed the 105°F requirement.

- During the Recirculation cases there are no RCPs running so the limiting temperature is the CCWS temperature of 200°F.

- For the Refueling case, subsequent evaluation by CP&L and Raytheon has indicated that higher CCW supply temperatures are acceptable.

All component outlet temperatures are sufficiently below the CCWS design temperature of 200°F and are therefore acceptable. The results were provided to Raytheon for use in revised stress analyses of CCWS piping and supports (see the BOP Licensing Report).

4.1.6.5 Conclusions

The analysis described above represents a comprehensive reanalysis of the CCWS for HNP. The reanalysis evaluates the heat removal capability of the CCWS during all modes of plant operation

and demonstrates the projected adequacy of planned modifications to augment the flow capacity of the CCWS. In addition to the increased decay heat from the uprated conditions, the revised analysis evaluated the effects of new spent fuel pool heat loads, CCW and RHR heat exchanger UAs, and new CCW and SW flow rates for each of 10 plant configurations. This analysis demonstrates the ability of the HNP CCW system with proposed modifications to provide the required cooling capacity to the components serviced by the CCW system, based on not only the uprated power level, but also the current plant parameters. The analysis at the NSSS power uprating of 2912.4 MWt with the replacement steam generators bounds plant operation at the current power level of 2787.4 MWt with the replacement steam generators.

Table 4.1.6-1 Components Serviced by CCW
Letdown Heat Exchangers
Excess Letdown Heat Exchanger
Seal Water Heat Exchanger
Reactor Coolant Pump Thermal Barrier Heat Exchanger
Reactor Coolant Pump Upper and Lower Bearing Oil Coolers
Reactor Coolant Drain Tank Heat Exchanger
Spent Fuel Pool Cooling Heat Exchangers
Residual Heat Exchangers
RHR Pump Seal Coolers
Gross Failed Fuel Detector
Sample Heat Exchanger Package

Table 4.1.6-2 Summary of CCW Performance For Plant Configurations			
Mode of Operation	**CCW Pump Flow Scenario**	**CCW Inlet Temperature (°F)**	**CCW Supply Temperature (°F)**
Plant Startup at 350°F	Min	115.7	106.4
	Max	115.3	106.8
Plant Startup at 557°F	Min	118.8	105.7
	Max	118.0	105.9
Normal Operations	Min	115.9	104.2
	Max	115.2	104.4
Shutdown at 350°F	Min	148.5	124.2
	Max	146.6	124.8
Shutdown at 140°F	Min	132.5	115.4
	Max	131.2	115.8
Refueling	Min	121.1	109.2
	Max	120.2	109.5
Safety Injection	Min	110.9	101.6
	Max	110.4	101.8
Recirculation at 243.5°F	Min	174.1	132.5
	Max	173.6	134.2
Recirculation at 200°F	Min	149.6	124.9
	Max	149.5	126.3
Loss of Offsite Power	Min	115.2	103.9
	Max	114.6	104.1

4.1.7 Gaseous Waste Processing System

This section provides a system review to evaluate the effects of the SGR/Uprating on the Gaseous Waste Processing System (GWPS).

4.1.7.1 System Function

The function of the GWPS is to collect, process and store fission gases that are contained in contaminated fluids within the plant resulting from normal plant operation, including anticipated operational occurrences. The release of gaseous effluents from the plant and expected offsite doses are as low as reasonably achievable (ALARA) and in conformance with the requirements of 10CFR20 and 10CFR50.

4.1.7.2 System Design Bases

The original design criteria of the GWPS is summarized as follows:

1. There will be no regularly scheduled discharge of radioactive fission gases from the GWPS to the atmosphere.

2. The flow of gases into the GWPS that cannot be processed and removed will be kept to an absolute minimum.

3. Fission gases will be removed from other primary plant systems to the extent possible and contained in the GWPS.

4. The accumulation of fission gases in the GWPS will not be permitted to create a radiation hazard either to plant operating personnel or to the general public.

5. In acknowledgment of the fact that it may become necessary to discharge waste gas at some time during the life of the plant, the system will include provisions to sample and control such discharges to ensure that releases are made within limits established by the applicable regulatory agencies. Provision will also be made to permit infrequent removal of small quantities of radioactive gases from the site in gas bottles or tanks.

6. The design of the GWPS is based on continuous plant operation assuming one percent failed fuel. This condition is assumed to exist over the life of the plant.

4.1.7.3 Evaluation of SGR/Uprating Effects

The processing load on the GWPS can be evaluated in terms of both the volumes of non-radioactive gases and the activities of radioactive fission gases that enter the system. The normal inflow to the GWPS is 0.7 SCFM from the Volume Control Tank (VCT) purge. During normal

plant operation, this flow consists of trace contamination of fission gases in a high purity stream of hydrogen. The GWPS can accommodate surges up to 1.7 SCFM such as may result from venting the Recycle Holdup Tank. No other anticipated operational occurrences will cause a significant surge in the GWPS process flow rate. The hydrogen is removed from this gas stream by the hydrogen recombiner, leaving the fission gases to be collected and stored in the Gas Decay Tanks. No change to the normal input flow rate to the GWPS has been identified as a result of the SGR/Uprating for HNP.

The HNP Gaseous Waste Processing System (GWPS) has the capability for long-term storage of radioactive gases generated in the RCS, eliminating the need for frequent routine gaseous discharges and allowing discharges to be made under favorable environmental conditions. The original sizing and number of Gas Decay Tanks (GDTs) for the HNP GWPS was based on a twin-unit station design, in which the GWPS was shared by the two units, and on distributing the activity among the tanks so as to minimize the dose associated with a GDT rupture. The HNP GWPS has the same number of gas decay tanks (ten) and capacity (600 cu. ft. each) as that associated with a twin-unit station. The standard Westinghouse design for a single unit 3-loop plant, such as HNP, includes only seven tanks. Furthermore, the updated waste gas plant inventories for HNP documented in the revised Radiation Analysis Manual (Reference 1) are found to be in the range of 1.5 to 3.7 times lower than that associated with a twin-unit, 4-loop plant design that includes ten gas decay tanks.

The ability of the GWPS to meet the design bases was demonstrated in Chapter 15 of the HNP FSAR, which documents that doses for the GDT rupture accident were calculated to be less than 5% of the applicable 10CFR100 limits. Note that the GDT rupture analysis reported in the FSAR is not based on the system design basis of distributing activity among the tanks but, instead, conservatively assumes that the primary coolant noble gas inventory is transferred to one GDT (with decay taken into account during the coolant degassing process). Note also that the GDT rupture and dose consequence analysis described in the FSAR was performed at a core power level of 2900 MWt, which is the same as the uprated power level.

Westinghouse has determined that the noble gas inventory for the HNP (as dose equivalent Xe-133) in the plant at shutdown for the uprated conditions is not significantly different than that associated with source terms previously provided by Westinghouse in Reference 2. Based on the assumption that the dose values reported in the FSAR are proportional to the dose equivalent Xe-133 at shutdown, it is concluded that the offsite doses remain less than 5% of the limits as reported in the FSAR. Thus, substantial margin remains in the design of the HNP waste gas system with respect to the 10CFR100 limits for offsite radiological consequences of the postulated GDT rupture. From this evaluation it is concluded that the existing waste gas system design of the HNP remains adequate for the waste gas activity inventories associated with the SGR/Uprating conditions.

4.1.7.4 Conclusion of System Review

This evaluation of the HNP Gaseous Waste Processing System has described the system design bases and functional requirements. The bases for the system performance in ensuring that release of gaseous effluents from the plant and expected offsite doses are as low as reasonably achievable (ALARA) and in conformance with the requirements of 10CFR20 and 10CFR50 were also described. On the basis of this evaluation, it can be concluded that the design of the HNP GWPS remains capable of meeting the design basis functional requirements and performance criteria of the system. The GWPS system supports operation at the current power with steam generator replacement as well as steam generator replacement in conjunction with power uprate.

4.1.7.5 References

1. WCAP-15397, "Radiation Analysis Manual, Model 312, Carolina Power and Light Company, Shearon Harris Nuclear Power Plant, Unit 1," April 2000.

2. Westinghouse "Radiation Analysis Manual, Model 312, Carolina Power and Light Company, Shearon Harris Nuclear Power Plant, Units 1, 2, 3 and 4," Rev. 2, December 1978.

4.1.8 Liquid Waste Processing System

This section provides a system review to evaluate the effects of the SGR/Uprating on the Liquid Waste Processing System (LWPS).

4.1.8.1 System Function

The function of the LWPS is to collect, store, process and control the release of radioactive and potentially radioactive liquids associated with operation, refueling and maintenance of the plant. The discharge of treated wastes is controlled and monitored to ensure that any discharges are as low as reasonably achievable (ALARA) and that they are in conformance with the requirements of 10CFR20 and 10CFR50.

4.1.8.2 System Design Bases

The original design basis of the Liquid Waste Processing System is summarized as follows:

1. Fission product concentrations in the reactor coolant are based on operation with cladding defects in one percent of the fuel assemblies.

2. Reactor coolant leakage into the system that cannot be recycled is assumed to be 20 gallons per day at ambient temperature.

3. Non-reactor grade water which enters the system in such a manner as to mix with the reactor coolant leakage is assumed to be 80 gallons (leakage and lab rinses) per day at ambient temperature.

4. Based on normal plant conditions, the system shall be capable of treating the combined leakage so that the final discharge will be less than one percent of the maximum permissible concentrations (MPC) as outlined in 10CFR20 for identified isotopes and 10 percent of MPC for isotopes released on an unidentified basis at the time of release.

5. Laundry, hot shower and decontamination wastes are processed before discharging.

6. A total design overall decontamination factor (DF) of $2x10^9$ is assumed for estimating isotopic releases.

4.1.8.3 Evaluation of SGR/Uprating Effects

The processing load on the LWPS can be evaluated in terms of both the volume of liquid waste that the system must process and the radionuclide activity that is present in the waste. Table 11.2.1-1 of the HNP FSAR also contains projected expected annual and daily average inputs to the LWPS subsystems from various sources within the plant. Because no equipment

change is anticipated to be required to implement the steam generator replacement or power uprating, the volumes of liquid wastes from these sources also are not expected to change.

The LWPS limits release of liquid radwaste to the environs to meet the as low as reasonably achievable (ALARA) criteria. The evaluation of releases of radioactive materials is based on the volumes of radwaste, as described above, and radionuclide activities and decontamination factors documented in Sections 11.2 and 11.3 of the HNP FSAR. It is noted that the design basis fission and corrosion product specific activities contained in Table 11.1.1-1 of the HNP FSAR are based on a core thermal power rating of 2900 MWt, which equals the uprated plant condition.

Westinghouse has compared the fission product inventory from the updated Radiation Analysis Manual (Reference 1) to that listed in the original design basis for the LWPS (Reference 2). Some of the isotope activity concentrations are higher due to the plant uprating. A slight increase in RCS activity should result in a corresponding slight increase in waste holdup or floor drain tank contents activity due to RCS equipment drains and leakage. The original effluent calculations were based on nominal decontamination factors (DFs) for the processing equipment (e.g., 10 for demineralizers, 1000 for evaporators for all isotopes except tritium). The effect of the slight increase of RCS activity concentrations is judged to be within the accuracy of the assumed nominal DFs for the processing equipment. Therefore, the uprated fission product inventory remains within the design basis capability of the HNP LWPS.

4.1.8.4 Conclusion of System Review

This evaluation of the Liquid Waste Processing System in HNP has described the system design bases and functional requirements. The bases for the system performance in ensuring that any discharges are as low as reasonably achievable (ALARA) and in conformance with the requirements of 10CFR20 and 10CFR50 were also described. On the basis of this evaluation, it can be concluded that the design of the HNP LWPS remains capable of meeting the design basis functional requirements and performance criteria of the system. The LWPS system supports operation at the current power with steam generator replacement as well as steam generator replacement in conjunction with power uprate.

4.1.8.5 References

1. WCAP-15397, "Radiation Analysis Manual, Model 312, Carolina Power and Light Company, Shearon Harris Nuclear Power Plant, Unit 1," April 2000.

2. Westinghouse Systems Standard Design Criteria 1.1.8.1, "System Requirements and Equipment Sizing, Waste Processing System (Liquid Portion)," Rev. 0, October 1972.

4.1.9 Gross Failed Fuel Detector System

This section provides a system review to evaluate the effects of the SGR/Uprating on the Gross Failed Fuel Detector System (GFFDS).

4.1.9.1 System Function

The GFFDS monitors the delayed neutron activity from fission products in the reactor coolant to provide an alarm indication of possible gross fuel failure. The GFFDS monitors the delayed neutron activity in a continuous fluid sample drawn from the hot leg in either Loop 2 or Loop 3 of the RCS. The GFFDS is a reliable indicator of fuel failure only when the reactor is at power.

4.1.9.2 System Design Bases

The original design basis of the Gross Failed Fuel Detector System is summarized as follows.

The neutron activity in the core arises from the delayed neutron emitting fission products that occur in the core, primarily Br-87, Br-88 and I-137. Consistent with the half-lives of these isotopes, the required sample delay time is approximately 53 seconds measured from the center of the core to the neutron detector in the GFFDS. A minimum of 28 seconds delay is maintained in the GFFDS sample line inside containment to meet N-16 decay requirements. The GFFDS sample flow rate is controlled within the range of 0.34 to 0.44 gpm to meet the delay time requirements described above.

4.1.9.3 Evaluation of SGR/Uprating Effects

The implementation of the steam generator replacement or the power uprating will have no effect on the design bases of the GFFDS. The critical isotopes that indicate fuel failure and their half-lives will remain the same, as will the required delay time in the sample line from the center of the core to the GFFDS detector and the required delay time inside containment. Consequently, the system design and operating parameters of the GFFDS are not expected to change when the SGR/Uprating conditions are put into effect.

4.1.9.4 Conclusion of System Review

This evaluation of the Gross Failed Fuel Detector System in HNP has described the system design bases and functional requirements. The bases for the system performance in monitoring the reactor coolant for fission products indicative of gross fuel failures also were described. On the basis of this evaluation, it can be concluded that the design of the HNP GFFDS remains capable of meeting the design basis functional requirements and performance criteria of the system for the conditions associated with operation of the current power and steam generator replacement as well as steam generator replacement in conjunction with power uprating.

4.2 NSSS/BOP Fluid Systems Interfaces

4.2.1 Introduction

As part of the Harris Nuclear Plant (HNP) Steam Generator Replacement/Uprating (SGR/Uprating) Project, the following Balance-of-Plant (BOP) fluid systems were reviewed to assess compliance with Westinghouse Nuclear Steam Supply Systems (NSSS)/BOP interface requirements:

- Main Steam System
- Steam Dump System
- Condensate and Feedwater System
- Auxiliary Feedwater System
- Steam Generator Blowdown System

The various interface systems were reviewed with the purpose of providing interface information which could be used in the more detailed BOP analyses. The results of those analyses are provided in the BOP Licensing Report.

The review was performed based on the range of NSSS design parameters developed to support steam generator replacement coupled with an uprated NSSS power level of 2912.4 MWt. The eight sets of approved design parameters, shown in Section 2, form the design bases for the project. In addition, 32 cases of best estimate parameters were generated to predict actual plant operating conditions. The 32 best estimate cases provide predictions of how the plant may actually operate. The major difference between design and best estimate parameters are the assumptions with respect to core flow and the steam generator tube fouling. If the range of best estimate parameters versus the range of design parameters provided a more conservative basis for a specific NSSS/BOP interface evaluation, the more conservative best estimate parameters were assumed.

This evaluation was focused on the uprated NSSS power of 2912.4 MWt. However, a review was performed considering the replacement steam generators and the current NSSS power of 2787.4 MWt.

4.2.2 Description of Analyses and Evaluation

Evaluations of the above BOP systems relative to compliance with Westinghouse NSSS/BOP interface guidelines were performed to address design and best estimate parameters for SGR/Uprating analyses, which include ranges for parameters such as T_{avg} (572° to 588.8°F) and steam generator tube plugging (0 to 10 percent average). These ranges of NSSS design and best estimate parameters result in ranges of BOP parameters such as steam generator outlet steam

pressure (824 to 1011 psia). In addition the parameters include a feedwater temperature range of 375° to 440°F. The NSSS/BOP interface evaluations were performed to address these ranges of NSSS and BOP parameters.

A comparison of the SGR/Uprating design parameters with the original Westinghouse Performance Capability Working Group (PCWG) parameters previously evaluated for systems and components indicates differences that could impact the performance of the above BOP systems. For example, the increase in NSSS power of approximately 4.6 percent (to 2912.4 MWt) and the upper limit of T_{avg} (588.8°F) would result in about a 5.3 percent increase in steam/feedwater mass flow rates. Additionally, the average steam generator tube plugging level of 10 percent in combination with the lower limit of T_{avg} (572°F) would result in a reduction in full-load steam pressure from 964 to 824 psia.

4.2.3 Acceptance Criteria

The NSSS/BOP system interface requirements are delineated in the Westinghouse Steam Systems Design Manual. For each BOP System evaluated, the specific NSSS/BOP system interface requirements are delineated in subsection 4.2.4.

4.2.4 Results

The results of the NSSS/BOP interface evaluations are delineated below.

4.2.4.1 Main Steam System

The uprating coupled with the potential reduction in full-load steam pressure to the average minimum value of 824 psia impacts main steam line pressure drop. At the average minimum steam generator pressure of 824 psia, the full-load steam mass flow rate would increase about 4.6 percent; however, due to the reduced operating pressure and the lower-density steam, the volumetric flow rate would increase by approximately 24.1 percent and steam line pressure drop would increase by approximately 29.8 percent. Based on the range of NSSS design parameters for SGR/Uprating to 2912.4 MWt, the lowest steam generator pressure would result in a pressure at the turbine inlet valves of approximately 784 psia.

The following summarizes the Westinghouse evaluation of the major steam system components relative to the SGR/Uprating conditions. The major components of the Main Steam System (MSS) are the Steam Generator Main Steam Safety Valves (MSSVs), the main steam Power Operated Relief Valves (PORVs), and the Main Steam Isolation Valves (MSIVs).

4.2.4.1.1 Steam Generator Main Steam Safety Valves

The setpoints of the MSSVs are determined based on the design pressure of the MSS (1185 psig) and the requirements of the ASME Boiler and Pressure Vessel (B&PV) Code. Since the design pressure of the MSS has not changed with the SGR/Uprating, there is no need to revise the setpoints of the safety valves.

The MSSVs must have sufficient capacity so that main steam pressure does not exceed 110 percent of the MSS design pressure (the maximum pressure allowed by the ASME B&PV Code) for the worst-case loss-of-heat-sink event. Based on this requirement, Westinghouse applies the conservative criterion that the valves should be sized to relieve 105 percent of the maximum calculated steam flow at an accumulation pressure not exceeding 110 percent of the MSS design pressure.

The Harris Nuclear Plant has 15 safety valves with a total capacity of 13.56×10^6 lb/hr, which provides about 105.3 percent of the maximum calculated steam flow of 12.87×10^6 lb/hr for the uprating. Therefore, based on the range of PCWG NSSS parameters for the SGR/Uprating, the capacity of the installed MSSVs meets the Westinghouse sizing criterion.

Additionally, the capacity of an individual MSSV (as well as each PORV and steam dump valve) is limited to 970,000 lb/hr at 1200 psia. Since the actual capacity of any single safety valve is about equal to 964,119 lb/hr at 1200 psia, the maximum capacity criteria is satisfied.

4.2.4.1.2 Main Steam Power Operated Relief Valves

The PORVs, which are located upstream of the MSIVs and adjacent to the MSSVs, are automatically controlled by steam line pressure during plant operations. The PORVs automatically modulate open and exhaust to the atmosphere whenever the steamline pressure exceeds a predetermined setpoint to minimize safety valve lifting during steam pressure transients. As the steamline pressure decreases, the PORVs modulate closed and reseat at a pressure below the opening pressure. The PORV set pressure for these operations is between zero-load steam pressure and the setpoint of the lowest-set MSSVs. As neither of these pressures change for the proposed range of NSSS design parameters, there is no need to change the PORV setpoint.

The primary function of the PORVs is to provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere when either the condenser, the condenser circulating water pumps, or steam dump to the condenser is not available. Under such circumstances, the PORVs in conjunction with the Auxiliary Feedwater System (AFWS) permit the plant to be cooled down from the pressure setpoint of the lowest-set MSSVs to the point where the Residual Heat Removal System (RHRS) can be placed in service. During cooldown, the PORVs are either automatically or manually controlled. In automatic, each PORV

proportional and integral (PI) controller compares steam line pressure to the pressure setpoint, which is manually set by the plant operator.

In the event of a tube rupture event in conjunction with loss of offsite power, the PORVs are used to cool down the Reactor Coolant System (RCS) to a temperature that permits equalization of the primary and secondary pressures at a pressure below the lowest-set MSSV. RCS cooldown and depressurization are required to preclude steam generator overfill and to terminate activity release to the atmosphere.

Each steam generator PORV is required to have a capacity at least equal to 64,000 lb/hr at 100-psia inlet pressure. At uprated power, this capacity permits a plant cooldown rate of 50°F/hr to RHRS operating conditions assuming a minimum of two hours at hot standby. This sizing is compatible with normal cooldown capability and minimizes the water supply required by the AFWS. This is based on one train of auxiliary feedwater (AFW) operating and flow going through two steam generators. The design capacity of the installed PORVs is 65,464 lb/hr/valve (full open) at 100 psia. Therefore the capacity of the installed PORVs meets the Westinghouse sizing criterion.

4.2.4.1.3 Main Steam Isolation Valves and Main Steam Isolation Bypass Valves

The MSIVs are located outside the containment and downstream of the MSSVs. The valves function to prevent the uncontrolled blowdown of more than one steam generator and to minimize the RCS cooldown and containment pressure to within acceptable limits following a main steamline break. To accomplish this function, the original design requirements specified that the MSIVs must be capable of closure within 5 seconds of receipt of a closure signal against steam break flow conditions in either the forward or reverse direction.

Rapid closure of the MSIVs following a postulated steam line break causes a significant differential pressure across the valve seats and a thrust load on the main steam system piping and piping supports in the area of the MSIVs. The worst cases for differential pressure increases and thrust loads are controlled by the steam line break area (i.e., mass flow rate and moisture content), the throat area of the steam generator flow restrictors, valve seat bore, and no-load operating pressure. As these variables and no-load operating pressure are not impacted by the SGR/Uprating, the design loads and associated stresses resulting from rapid closure of the MSIVs will not change. Consequently, SGR/Uprating has no significant impact on the interface requirements for the MSIVs.

The MSIV bypass valves are used to warm up the main steam lines and equalize pressure across the MSIVs prior to opening the MSIVs. The MSIV bypass valves perform their function at no-load and low-power conditions where SGR/Uprating has no significant impact on main steam conditions (e.g., steam flow and steam pressure). Consequently, SGR/Uprating has no significant impact on the interface requirements for the MSIV bypass valves.

4.2.4.1.4 Main Steam System Conclusions

The capacity of the installed MSSVs meets the Westinghouse sizing criterion for the proposed range of NSSS operating conditions.

The capacity of the installed main steam PORVs meets the Westinghouse sizing criterion for the proposed range of NSSS operating conditions.

The MSIVs and MSIV bypass valves are not adversely impacted by the SGR/Uprating.

4.2.4.2 Steam Dump System

The Steam Dump System creates an artificial steam load by dumping steam from ahead of the turbine valves to the main condenser. The Westinghouse sizing criterion recommends that the steam dump system (valves and pipe) be capable of discharging 70 percent of the rated steam flow at full-load steam pressure to permit the NSSS to withstand an external load reduction of up to 100 percent of plant rated electrical load without a reactor trip. To prevent a trip, this transient requires all NSSS control systems to be in automatic, including the Reactor Control System, which accommodates about 20 percent of the load reduction. In addition, a steam dump capacity of just 40 percent of rated steam flow at full-load steam pressure prevents MSSV lifting following a reactor trip from full power.

4.2.4.2.1 Steam Dump System Major Components

The Harris Nuclear Plant is provided with six condenser dump valves and eight atmospheric dump valves. The total capacity for all 14 valves provides a steam dump capability of greater than 70 percent of the original maximum guaranteed steam flow with one valve out of service.

NSSS operation within the range of design parameters at lower steam generator pressures and higher steam flows will result in a reduced steam dump capability. An evaluation for the SGR/Uprating indicates that the total steam dump capacity with 14 valves in service could be as low as 68.7 percent of rated steam flow (12.76×10^6 lb/hr), or 8.77×10^6 lb/hr, at a full-load steam pressure equal to 825 psia. These operating conditions are based on an NSSS power level of 2912.4 MWt, an average steam generator tube plugging (SGTP) level of 10 percent, and a T_{avg} in the lower end of the operating range (572°F). Therefore, the calculated steam dump capacity is 68.7 percent versus the Westinghouse recommended capacity of 70 percent for 100 percent load rejection. The NSSS control systems analysis in Section 4.3 provides a further evaluation of the Steam Dump System adequacy at the SGR/Uprating conditions.

Note that at the upper end of the T_{avg} operating range and a full-load best estimate steam generator pressure of 1011 psia, steam dump capacity is about 84.9 percent of rated flow (11.8×10^6 lb/hr), or 10.02×10^6 lb/hr with 13 valves in service.

To provide effective control of flow on large step load reductions or plant trip, the steam dump valves are required to go from full-closed to full-open in 3 seconds at any pressure between 50 psi less than full load pressure and steam generator design pressure. The dump valves are also required to modulate to control flow. Positioning response may be slower with a maximum full stroke time of 20 seconds. These requirements are still applicable for the NSSS operating conditions for the SGR/Uprating.

4.2.4.2.2 Steam Dump System Conclusions

The conclusion of the assessment of the uprated conditions on the Steam Dump System is that for the range of operating conditions for the uprating, steam dump capacity is less than the Westinghouse recommended capacity. The design load rejection capability has been confirmed by a control systems operability assessment (i.e., margin to trip analysis), and the condenser Steam Dump System has been shown to be acceptable.

4.2.4.3 Condensate and Feedwater System

The Condensate and Feedwater System (C&FS) must automatically maintain steam generator (SG) water levels during steady-state and transient operations. The range of NSSS design parameters will impact both feedwater volumetric flow and system pressure drop. The volumetric flow may increase by as much as 5.8 percent or decrease by as much as 8.4 percent. Therefore, system pressure drop may increase by as much as 11.4 percent or decrease by as much as 12.0 percent respectively during full-power operation. Also, a comparison of the uprated design and best estimate parameters with the original design parameters indicates that the SG full-power operating steam pressure may increase by as much as 47 psi (964 to 1011 psia) or decrease by as much as 139 psi (964 to 825 psia).

The pre-heater bypass lines and associated valves and the feedwater isolation valve bypass lines and associated valves will not be required with the replacement steam generators (RSGs). The general arrangement and layout interface requirements that pertain to the C&FS are provided in the Westinghouse Steam Systems Design Manual and these requirements are not impacted by the proposed change in NSSS operating conditions.

The major components of the C&FS are the Feedwater Isolation Valves (FIVs), the Feedwater Control Valves (FCVs), Feedwater Bypass Control Valves (FBCVs), and the C&FS Pumps.

4.2.4.3.1 Feedwater Isolation, Feedwater Control and Feedwater Bypass Control Valves

The FIVs are located outside containment and downstream of the FCVs and FBCVs. The valves function in conjunction with the primary isolation signals to the FCVs, FBCVs and backup trip signals to the feedwater pumps to provide redundant isolation of feedwater flow to the steam generators following a steam line break or a malfunction in the steam generator level control system. Isolation of feedwater flow is required to prevent containment overpressurization and

excessive RCS cooldowns. To accomplish this function, the FIVs and the backup FCVs and FBCVs must be capable of fast closure, following receipt of any feedwater isolation signal.

The quick-closure requirements imposed on the FIVs, and the backup FCVs and FBCVs cause dynamic pressure changes that may be of a large magnitude and must be considered in the design of the valves and associated piping. The worst loads occur following a steam break from no-load conditions with the conservative assumption that all feedwater pumps are in service providing maximum flow following the break. Maximum feedwater flow may be impacted by the proposed feedwater system design changes that include an increase in feedwater pump head. Further evaluation of the C&FS, including the feedwater system design changes, is contained in the BOP Licensing Report.

4.2.4.3.2 Condensate and Feedwater System Pumps

The C&FS available head in conjunction with the FCV/FBCV characteristics must provide sufficient margin for feed control to ensure adequate flow to the steam generators during steady-state and transient operation. A continuous steady feed flow should be maintained at all loads. Further evaluation of the C&FS, including the feedwater and condensate pumps, is contained in the BOP Licensing Report.

4.2.4.3.3 Condensate and Feedwater System Pumps Conclusions

The evaluations of the Condensate and Feedwater System at the uprated conditions show that the hydraulics of the C&FS should permit operation over the entire range of full power NSSS operating conditions for the uprating. Further evaluation of the C&FS is contained in the BOP Licensing Report.

4.2.4.4 Auxiliary Feedwater System

The AFWS supplies feedwater to the secondary side of the steam generators at times when the normal feedwater system is not available, thereby maintaining the heat sink of the steam generators. The system provides feedwater to the steam generators during normal unit startup, hot standby, and cooldown operations and also functions as an Engineered Safeguards System. In the latter function, the AFWS is directly relied upon to prevent core damage and system overpressurization in the event of transients and accidents such as a loss of normal feedwater or a secondary system pipe break. The minimum flow requirements of the AFWS are dictated by accident analyses, and since the power uprating impacts these analyses, evaluations of the limiting transients and accidents must be performed to confirm that the AFWS performance is acceptable at the uprated conditions.

4.2.4.4.1 Auxiliary Feedwater Storage Requirements

The AFWS pumps are normally aligned to take suction from the condensate storage tank (CST). To fulfill the Engineered Safety Features (ESF) design functions, sufficient feedwater must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition.

The limiting transient with respect to CST safety-related inventory requirements is the loss-of-offsite power (LOOP) transient. The HNP licensing basis dictates that in the event of a LOOP, sufficient CST useable inventory must be available to bring the unit from full-power to hot-standby conditions, maintain the plant at hot standby for 12 hours, and then cool down the RCS to the RHRS cut-in temperature (350°F) in 4 hours. In light of these design bases requirements, the analysis-of-record concluded that a minimum useable inventory of 233,726 gal. is required in the CST. The design of the CST permits a maximum safety-related contained inventory of 270,000 gal. and a maximum safety-related useable inventory of about 235,000 gal. The plant Technical Specifications require a safety-related contained inventory of 270,000 gal.

A revised analysis for the range of NSSS design parameters approved for the SGR/Uprating indicates that 238,048 gal. is required to satisfy the present licensing bases: 12-hr hot standby and 4-hr cooldown time to an RHRS cut-in temperature of 350°F.

Further evaluation of the AFW system is contained in the BOP Licensing Report.

4.2.4.4.2 Auxiliary Feedwater System Conclusions

The minimum flow requirements of the AFWS are dictated by accident analysis and since the uprating impacts these analyses, evaluations of the limiting transients and accidents have been performed to confirm that the AFWS performance is acceptable at the uprated operating conditions. These analyses are described in Section 6.0 of this report.

Further analysis of CST minimum inventory requirements for the SGR/Uprating operating conditions is contained in the BOP Licensing Report.

4.2.4.5 Steam Generator Blowdown System

The Steam Generator Blowdown System (SGBS) is used in conjunction with the Chemical Feed and Sampling Systems to control the chemical composition of the steam generator shell water within the specified limits. The blowdown system also controls the buildup of solids in the steam generator water. The RSGs are provided with two blowdown connections, a tube sheet connection and a shell connection. The tube sheet connection is required to control the buildup of particulate matter that enters the steam generator and tends to accumulate by settling in the tube sheet area. Either the tube sheet connection or the shell connection or both can be used to control dissolved solids and maintain chemistry within specifications.

The required blowdown flow rates during plant operation will not be significantly impacted by power uprate, since neither the rate of addition of dissolved solids or the rate of addition of particulates into the steam generators will be significantly impacted by the SGR/Uprating.

A hydraulic analysis was performed to ensure compatibility of the blowdown system control valves with the piping changes dictated by the RSGs and approved a range of NSSS operating conditions. This work also addressed a SGBS design change to locate several valves from inside the secondary shield wall to a location outside the wall. The following summarizes the results of the analysis:

- The pipe re-route required to install the RSGs and to move the system valves from inside to outside the secondary shield wall is acceptable in terms of system hydraulics.

- The required split in tube sheet blowdown versus shell blowdown is compatible with both the hydraulic design of the RSGs and the re-route of the blowdown system pipe.

- The size of the trim in the SGBS flow control valves is adequate to establish and control a maximum blowdown flow of 100,000 lb/hr over the allowable range of steam generator shell side operating pressures.

4.2.4.5.1 Steam Generator Blowdown System Conclusions

The actual required blowdown flow rates during plant operation will not be significantly impacted by the SGR/Uprating, since neither the rate of addition of dissolved solids or the rate of addition of particulates into the steam generators will be significantly impacted by the power uprating.

Further evaluation of the SGBS is contained in the BOP Licensing Report.

4.2.5 NSSS/BOP Fluid Systems Interfaces Conclusion

In general, these evaluation results are applicable to the SGR at the uprated power of 2912.4 MWt.

HNP operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt was also reviewed. The review confirmed that with one exception, the analyses and evaluations performed at the uprated NSSS power of 2912.4 MWt are bounding for HNP operation with the replacement steam generators at the current NSSS power. The one exception is the analysis and results presented for the main feedwater control valve. The FCV evaluation performed for NSSS operation at the SGR/Uprating conditions is not bounding for NSSS operation with the RSGs at the current power level of 2787.4 MWt. For NSSS operation at SGR/current power conditions, the minimum FCV lift is expected to vary an additional amount (about 5 percent). This small change in expected valve lift is not significant and should not be a

problem in terms of plant operability. Refer to the BOP Licensing Report for additional information.

4.3 NSSS Control Systems

4.3.1 Introduction

As the proposed plant conditions with the HNP steam generator replacement (SGR) and uprated power level are somewhat different from those assumed in the original design, it was necessary to reexamine the control systems, their setpoints, and the sizing of the critical NSSS components. Therefore, the following evaluations were performed for the NSSS control systems:

- Control Systems Stability Analysis
  - Rod Control System
  - Steam Dump Control System
- Plant Operability Margins Evaluation
- NSSS Component Sizing
  - Power Operated Relief Valves
  - Pressurizer Spray Valves
  - Pressurizer Heaters
  - SG Main Steam Safety Valves
  - Pressurizer Safety Valves
- Steam Generator Level Control System Evaluation
- Low Temperature Over Pressurization System Evaluation

The following sections summarize the main results and conclusions of this evaluation.

4.3.1.1 Control Systems Stability Analysis

The NSSS Control Systems are designed to provide a stable and acceptable response to normal condition transients. An unstable control system may lead the plant parameters toward unbounded values and may cause excessive wear and tear on the control systems. In light of this, the stability of the Rod Control System and the Steam Dump Control System were evaluated for the HNP SGR/Uprating program.

Rod Control System

The results of the analysis showed acceptable stability in the Rod Control System for design basis 10-percent load increase, 5-percent ramp load increase, and large-load decrease transients at SGR/Uprating and current power, and beginning-of-life (BOL) and end-of life (EOL) conditions. High and low frequency rod stepping was not observed during the transient. The existing rod control system setpoints and time constant remain valid for the SGR/Uprating and current power conditions.

Steam Dump Control System

SGR and power uprating conditions have several effects on the Steam Dump Control Systems: (1) an uprated power ideally requires a larger steam dump capacity because of the higher steam flow, (2) a lower steam pressure reduces the effective steam dump capacity, and (3) the steam dump performance depends on the Steam Dump Control System setpoints, which are different at low and high T_{avg} conditions. Based on this, an evaluation of the Steam Dump Control System at HNP SGR/Uprating conditions was performed.

The evaluation showed that the plant response to normal condition transients was stable, but an automatic reactor trip occurred on the OTΔT trip setpoint on a full-load rejection transient. This is acceptable as full-load rejection capability is no longer an acceptance criterion for HNP due to balance-of-plant (BOP) system limitation at power uprate conditions. The plant response to a 50-percent load reduction from full-power transient was stable at both high and low T_{avg} and BOL and EOL conditions. The results of the turbine trip transient showed an acceptable controller response at high and low T_{avg} conditions. The steam dump system turbine-trip controller setpoint will provide adequate margin to the Engineered Safety Features (ESF) and Main Steam Safety Valve (MSSV) actuation setpoints following a turbine-trip transient.

In summary, performance of the control systems during both steady state and Condition I transients will not be degraded by the HNP SGR/Uprating conditions. The current control systems setpoints and time constants will remain applicable for the SGR/Uprating and current power conditions.

4.3.1.2 Plant Operability Margins Evaluation

The adequacy of available operating margin to various reactor trip and ESF actuation setpoints that are active during and following normal at-power (Condition I) operating transients at HNP SGR/Uprating conditions were evaluated. The Condition I transients considered for evaluation are initiated from full-power and, in some cases, reduced-power conditions.

There are 15 reactor trip and 10 ESF actuation signals active during at-power operation. The general warning alarm trip is also active during at-power operation. Other reactor trips are either manually actuated or are not active during at-power operation (e.g., high source

range/intermediate range Nuclear Instrumentation System trips, presence of a safety injection signal).

The evaluation demonstrated that sufficient margin exists to relevant reactor trips and ESF actuation setpoints during Condition I (normal operating) transients. An automatic reactor trip on OTΔT will occur on a full-load rejection (net-load trip) transient but adequate margin to OTΔT exists on a 50-percent load reduction transient. Since full-load rejection capability is no longer an acceptance criterion for HNP due to BOP system limitation, an automatic reactor trip on a full-load rejection transient is acceptable. The control and protection current and revised setpoints for the SGR/Uprating will provide adequate operating margin during Condition I transients.

The results obtained with the Model Δ75 replacement steam generators at the uprated NSSS power of 2912.4 MWt are also applicable for the SGR at the current NSSS power of 2787.4 MWt.

4.3.1.3 NSSS Component Sizing Evaluation

The HNP SGR/Uprating nominal full-power design conditions are significantly different from those assumed in the original sizing calculations to establish the minimum sizing requirements for the following NSSS components. Therefore, the installed capacities of these components were evaluated at the SGR/Uprating conditions.

- Pressurizer Power Operated Relief Valves
- Pressurizer Spray Valves
- Pressurizer Heaters
- SG Main Steam Safety Valves
- Pressurizer Safety Valves

For comparison purposes, the sizing analyses for the pressurizer and steam generator safety valves were also performed at HNP SGR/Uprating conditions. The adequacy of the installed pressurizer and steam generator safety valves is assessed by the appropriate safety analyses in Section 6.2, Non-LOCA Transients.

Pressurizer PORVs

The sizing basis for the pressurizer Power Operated Relief Valves (PORVs) is to prevent the pressurizer pressure from reaching the high pressurizer pressure reactor trip setpoint for the design basis full-load rejection with steam dump transient. This criterion is conservatively met if the total PORV capacity is greater than or equal to the peak pressurizer in-surge flow rate during and following this transient. The PORV opening characteristics are not modeled in the sizing

analysis to prevent the valve characteristics from interfering with the peak surge-rate calculation. The effect of PORV opening characteristics on peak pressurizer pressure is evaluated in the control system analysis.

The total capacity of the installed pressurizer PORVs is higher than the required capacity at SGR/Uprating and SGR/current power conditions and is, therefore, acceptable for SGR/Uprating and SGR/current power conditions.

Pressurizer Spray Valves

The sizing basis for Pressurizer Spray Valves is to prevent challenges to the pressurizer PORVs from a 10-percent step-load decrease transient. For load decrease amounts of up to 10-percent power, the spray valves are the sole means of providing pressure control without actuating the pressurizer PORV in the automatic mode of pressure control.

The total capacity of the installed pressurizer spray valves is acceptable for SGR/Uprating and SGR/current power conditions.

Pressurizer Heaters

The Pressurizer Heater total capacity is proportional to the pressurizer volume. For Westinghouse plants, the required capacity for pressurizer heaters is one kilowatt per one cubic foot of pressurizer total volume. With a nominal 1400 ft^3 pressurizer at HNP, this translates to pressurizer heater total capacity of 1400 kW. The total capacity of the installed heaters is 1400 kW and, therefore, meets the Westinghouse standard requirements of 1.0 kW/ft^3 of pressurizer total volume. The adequacy of the pressurizer heater installed capacity at SGR/Uprating conditions was verified, based on analysis of a 10-percent load change and full-load rejection transients.

The total installed heater capacity is acceptable for SGR/Uprating and SGR/current power conditions.

Steam Generator Main Steam Safety Valves

The sizing basis for steam generator MSSVs is to limit the pressure in the steam generator so it does not exceed 110 percent of the steam generator design pressure on a complete loss-of-load transient. Westinghouse applies the conservative criterion that the safety valves should be sized to relieve 105 percent of the maximum calculated steam flow at an accumulation pressure not exceeding 110 percent of the main steam system design pressure.

The installed steam generator MSSV total capacity at each valve's respective setpoint plus 3-percent accumulation will limit the steam generator pressure below 110 percent of the steam generator design pressure on a sizing design basis transient at SGR/Uprating and SGR/current

power conditions. Contingent upon meeting the appropriate safety analyses criteria, the installed MSSVs are acceptable at SGR/Uprating and SGR/current power conditions.

Pressurizer Safety Valves

The sizing basis for the pressurizer safety valves is designed to limit the pressurizer pressure not to exceed 110 percent of the RCS design pressure on a complete loss-of-load transient. This criterion is conservatively met if the total capacity of the pressurizer safety valves is greater than or equal to the peak pressurizer in-surge flow rate during and following this transient.

The total capacity of the installed pressurizer safety valves at setpressure plus 3-percent accumulation is higher than the calculated maximum required capacity at SGR/Uprating and SGR/current power conditions. Contingent upon meeting the appropriate safety analysis criteria, the installed pressurizer safety valves are acceptable at SGR/Uprating and SGR/current power conditions.

4.3.1.4 Steam Generator Level Control System Evaluation

The performance of the Steam Generator Level Control System (SGLCS) has been evaluated as part of the HNP SGR/Uprating program. The predicted responses will allow plant personnel to anticipate margins to level setpoints and deviation alarm limits and can be used in preparation of plant test procedures or training. The predicted responses have also been evaluated to determine some acceptable combinations of SGLCS setpoints that are based on the simulation results. It is intended, however, that both the simulation results and the plant test and operating data, taken together, be considered in determining the final SGLCS setpoint values.

The current SGLCS setpoints will not provide acceptable control system response and margin to the steam generator narrow-range level (NRL) high-high trip setpoint on a large-load reduction transient. Acceptable SGLCS performance and margin to the NRL trip setpoints were obtained with the alternative (i.e., basically those originally implemented) SGLCS setpoints. Alternative sets of these setpoints are recommended based on the analysis results. The final selection of SGLCS setpoint values for operation should be determined based on the analytical results and recommendations, the results of transient testing during plant startup with the replacement steam generators, and plant performance during startup and operation. The results obtained with the Model Δ75 replacement steam generators at the uprated NSSS power of 2912.4 MWt are also applicable for the replacement steam generators at the current power of 2787.4 MWt.

4.3.1.5 Low Temperature Over Pressurization Transients Analyses

The HNP design basis Low Temperature Over Pressurization (LTOP), Mass Input (MI) and Heat Input (HI) transients were analyzed at SGR/Uprating conditions (LTOP transients are also known as Cold Overpressure Mitigation [COM] transients). The results may be used to determine the appropriate pressurizer PORV setpoint for low temperature operation. The design basis MI and

HI transients for HNP are an inadvertent start of one charging/safety injection pump (MI) and startup of a reactor coolant pump when the steam generator secondary side is 50°F hotter than the RCS primary side (HI). Results of the parameter sensitivity on pressurizer PORV characteristics may be used as necessary.

Using a Residual Heat Removal (RHR) relief valve for the LTOP event, the results indicated a peak RCS pressure of 490 psig for the MI transient and 557 psig (with RCS at 250°F) and 666 psig (with RCS at 300°F) for the HI transient. These results need to be modified by other considerations, such as static head elevation between RHR relief valve and RHR piping, for RHR system overpressure evaluation. The results of this analysis are used as input to subsequent calculations for LTOP transients. Refer to the BOP Licensing Report.

5.0 NSSS COMPONENTS

Evaluations were performed to assess the impact of the SGR/Uprating conditions on all the NSSS components. For most cases, the uprating-related input used for these evaluations were the PCWG parameters (given in Section 2), and the NSSS design transient changes (described in Section 3). In addition to these documents, the component designers also reviewed the original design information related to that component. The evaluations were performed to confirm that the components continue to satisfy the applicable codes, standards, and regulatory guides under the revised conditions.

Following is a list of the component areas that were assessed. A detailed discussion of each component is provided within the remainder of this section.

- Reactor Vessel Structural and Incore Instrumentation
- Reactor Vessel Integrity (Not Westinghouse Scope)
- Reactor Internals
- Fuel Assemblies Structural (Not Westinghouse Scope)
- Control Rod Drive Mechanisms Structural
- Reactor Coolant Loop Piping and Supports and Leak Before Break
- Reactor Coolant Pump and Motor
- Steam Generator Structural, Thermal/Hydraulic and Flow Induced Vibration
- Pressurizer Structural
- Fracture Integrity of NSSS Components
- NSSS Auxiliary Equipment Evaluation

5.1 Reactor Vessel

5.1.1 Reactor Vessel Structural Evaluation

5.1.1.1 Introduction

Evaluations were performed for the various regions of the Harris Nuclear Plant (HNP) reactor vessel to determine the stress and fatigue usage effects of NSSS operation at the revised operating conditions of the SGR/Uprating Project throughout the current plant operating license. The revised operating parameters in Section 2 identify vessel inlet and vessel outlet temperatures that define the steady state operating temperatures for the reactor vessel for a range of high and low temperature operation. The design transients for the reactor coolant system (RCS), in turn, define the temperature and pressure responses for a variety of transients that can occur during

either high temperature or low temperature operation. The revised transient temperature and pressure variations may affect both the maximum ranges of primary plus secondary stress intensity and the maximum cumulative fatigue usage factors for the reactor vessel. The evaluations assess the effects on the maximum ranges of stress intensity and fatigue usage factors at the most limiting locations in each of the regions of the reactor vessel as identified in the reactor vessel stress report and addendum (References 1 and 2).

5.1.1.2 Description of Analyses and Evaluations

The key evaluation input parameters are the revised NSSS parameters in Section 2, the NSSS component design transients, and the reactor vessel/reactor internals interface loads. The temperatures and pressures considered in the reactor vessel structural evaluation for the NSSS parameters are as follows:

Normal Operating Pressure:	2317 psig
Normal Operating Temperatures	
Vessel Inlet Temperature Range:	536.6°F to 557.4°F
Vessel Outlet Temperature Range:	604.2°F to 623.2°F
Zero Load Temperature:	557°F

Other input parameters include design inputs from the HNP reactor vessel equipment specification (Reference 3), revised reactor vessel nozzle and nozzle support pad loads (as necessary), and HNP design changes/modifications.

The revised RCS design transients for the HNP SGR/Uprating Project were reviewed and compared to the original design basis transients from Reference 3. This transient review determined which revised transients are more severe than their design basis counterparts by comparing rates, magnitudes and durations of the transient temperature variations as well as the magnitudes of the pressure variations. Based upon this review, a determination of which revised transients must be considered in the stress evaluations was made.

The transient review concluded that only four of the T_{hot} transient temperature variations were more severe for the SGR/Uprating than for the design basis. The more severe T_{hot} design transients include Loss of Power, Partial Loss of Flow, Control Rod Drop, and Inadvertent Safety Injection Actuation. In addition, RCS Cold Overpressurization was included as a design transient. The stress intensities for these revised transients were examined to determine their effect on the maximum ranges of stress intensity for the outlet nozzle. Calculations were performed to account for the changes in thermal stress due to the modified temperature variations.

The transient review also concluded that eight of the SGR/Uprating T_{cold} transient temperature variations were more severe than their design basis counterparts. The revised transients with more severe T_{cold} temperature variations include Plant Loading, Plant Unloading, Large Step Load Decrease, Loss of Load, Loss of Power, Reactor Trip from Full Power with No Cooldown, Control Rod Drop, and Inadvertent Safety Injection Actuation. RCS Cold Overpressurization was again considered as a design transient. These transients were applied to all regions of the reactor vessel pressure boundary other than the outlet nozzles, including the vessel inlet nozzles, main closure, bottom head, top head, main shell, and core support pads that are in contact with vessel inlet water (T_{cold}) during normal operation. The change in the transient thermal stress due to the revised and new transient temperature variations were calculated. The incremental thermal stress changes were then factored into the stress intensities, and the effects of the changes on the maximum ranges of stress intensity were evaluated.

In addition to the changes in the temperature variations, many of the revised design transients also exhibited changes in their RCS pressure variation. The incremental pressure stresses were calculated by scaling the original transient pressure stresses proportional to the change in the pressure variation, since pressure stress is directly proportional to pressure. The changes in pressure stress were also added into the revised stress intensities for both the T_{hot} and T_{cold} regions where appropriate. The revised transients that exhibited significant changes from the design basis transients in their pressure variations include Plant Loading, Plant Unloading, Loss of Power, Partial Loss of Flow, Reactor Trip with No Cooldown, Reactor Trip with Cooldown but No Safety Injection, Reactor Trip with Cooldown and Safety Injection, Control Rod Drop, and Inadvertent Safety Injection Actuation. The RCS pressure variation for RCS Cold Overpressurization was also considered. These revised pressure variations were considered in the evaluations of the various vessel regions even if the applicable transient temperature variation was unaffected.

Where appropriate, revised maximum ranges of stress intensity and maximum usage factors were calculated for the SGR/Uprating. In other cases, the original design basis stress analysis remains conservative so that no new calculations were necessary, and the maximum ranges of stress intensity and fatigue usage factors reported in the Chicago Bridge and Iron Co. stress report (Reference 1) and Westinghouse addendum (Reference 2) continue to govern.

Revised LOCA loads for the reactor vessel interfaces with the reactor internals at the main closure flange ledges, the outlet nozzle internal projections and the core support pads were developed. Also included were LOCA loads at the vessel/vessel support interface. These LOCA loads were compared to the faulted condition loads that were considered in References 1 and 2. The comparison revealed that only the LOCA loads at the core support pad interface with the internals lower radial keys exceeded the loads that were previously considered in the reactor vessel stress report. Therefore, a calculation was performed to determine the stresses at the core support pads. The combined normal plus LOCA stresses were then compared to the applicable faulted condition acceptance criteria.

The head adapter plugs were not previously included with the reactor vessel structural analysis because they were supplied separately from the reactor vessel and its associated equipment as defined in Reference 3. The head adapter plugs were also evaluated separately for the HNP SGR/Uprating. The evaluation was performed by applying a head adapter plug analysis that considered design transients that better match the SGR/Uprating transients in lieu of the original design analysis. The reference head adapter plug analysis was then reconciled with the HNP head adapter plug design including design transients, material, analytical methods and acceptance criteria. The reference analysis results were then applied to HNP.

5.1.1.3 Acceptance Criteria

The acceptance criteria for the reactor vessel structural analyses and evaluations are in accordance with the applicable requirements of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code with addenda through Winter 1971 (Reference 4). The applicable code for the head adapter plugs is the 1974 Edition of Section III of the ASME Boiler and Pressure Vessel Code (Reference 5). However, the applicable acceptance criteria are the same. The applicable acceptance criteria are as follows:

The maximum range of primary plus secondary stress intensity resulting from normal and upset condition design transient mechanical and thermal loads shall not exceed $3S_m$ at operating temperature, in accordance with Paragraph NB-3222.1 of Reference 4. The maximum cumulative usage factor resulting from the peak stress intensities due to normal and upset condition design mechanical and thermal loads shall not exceed 1.0, in accordance with the procedure outlined in Paragraph NB-3222.4 of Reference 4.

The faulted conditions shall meet the component criteria of Appendix F of the ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition (Reference 5: The 1974 Edition was used, since Appendix F was not yet included in the 1971 Edition). For the core support pad faulted condition analysis, the general primary membrane stress intensity limits are $2.4S_m$ for the Alloy 600 pads and $0.7S_U$ for the low-alloy steel vessel shell. The primary membrane plus bending limits are $3.6S_m$ for the pads and $1.05S_U$ for the vessel shell.

5.1.1.4 Results

The SGR/Uprating affects only one of the maximum ranges of stress intensity reported in the HNP reactor vessel stress report (References 1 and 2). The maximum range of stress intensity for the outlet nozzle safe end increased, but remains below the applicable ASME Boiler and Pressure Vessel Code, Section III limit. The evaluations show that for all other limiting locations, the existing design stress analyses remain conservative when the revised operating parameters and design transients are incorporated. The maximum cumulative fatigue usage factors at all of the limiting locations (except for the closure studs) increase somewhat from the values reported in Reference 2. However, the increases are all minimal, and all of the cumulative fatigue usage factors remain under the 1.0 limit with significant margin.

The updated maximum ranges of primary plus secondary stress intensity and maximum cumulative fatigue usage factors for the HNP reactor vessel accounting for the SGR/Uprating is shown in Table 5.1.1-1.

A comparison of the LOCA interface loads to the corresponding LOCA and faulted condition loads that were previously considered in References 1 and 2 concluded that only the core support pads required additional stress analysis to justify application of LOCA loads. LOCA loads at the core support pad interface were not included in the reactor vessel design as defined by Reference 3. The faulted condition calculations for the core support pads were performed to justify application of a total faulted condition core support pad load that included the total LOCA load. The results are summarized in Table 5.1.1-2.

5.1.1.5 Conclusions

Based upon the satisfactory results of the evaluations, the HNP reactor vessel is acceptable for plant operation at the revised conditions. Such operation of the reactor vessel is shown to be acceptable in accordance with the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code with Addenda through Winter 1971 (Reference 4) for the remainder of the plant license. Operation of the head adapter plugs is acceptable in accordance with the 1974 Edition of Section III of the ASME Boiler and Pressure Vessel Code with Addenda through Winter 1976 (Reference 5).

The results obtained with the Δ75 RSGs at the uprated power of 2912.4 MWt bound operation with the Δ75 RSGs at the current NSSS power of 2787.4 MWt.

5.1.1.6 References

1. Chicago Bridge and Iron Company Stress Report Certification and Final (Summary) Stress Report 157 "PWR Vessel Shearon Harris Nuclear Station-Unit 1, Westinghouse P.O. No. 150117," December 22, 1981.

2. Westinghouse Report and Certification MED-PCE-2537, "Addendum to Chicago Bridge and Iron Final Stress Report for the Shearon Harris Unit No. 1 Reactor Vessel," Winston K. Ma and John C. Schmertz, April 12, 1985. (Westinghouse Proprietary)

3. Westinghouse Equipment Specification 679119, Rev. 3, for the Shearon Harris Nuclear Station Unit No. 1 Reactor Vessel, Addendum to Equipment Specification 676413, Rev. 3, March 13, 1985 and Westinghouse Equipment Specification 676413, Rev. 3 for General Reactor Vessel, July 16, 1971. (Westinghouse Proprietary Class 2)

4. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," 1971 Edition with Addenda through Winter 1971, American Society of Mechanical Engineers, New York.

5. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," 1974 Edition with Addenda through Winter 1976, American Society of Mechanical Engineers, New York.

Table 5.1.1-1
Maximum Ranges of Primary Plus Secondary Stress Intensity and Maximum Cumulative Fatigue Usage Factors for the RV

Location	$P_L + P_b + Q$ Range	Previous	U_C	Previous
Outlet Nozzles	Nozzle: [ ]$^{(b,c)}$ < 3 S_m = 80.1 ksi Safe End: [ ]$^{(b,c)}$ < 3 S_m = 56.40 ksi	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$ < 1.0	[ ]$^{(b,c)}$
Inlet Nozzles	[ ]$^{(b,c)}$ < 3 S_m = 80.1 ksi	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$ < 1.0	[ ]$^{(b,c)}$
Main Closure Flange Region				
1. Closure Head Flange	[ ]$^{(b,c)}$ < 3 S_m = 80.1 ksi	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$ < 1.0	[ ]$^{(b,c)}$
2. Vessel Flange	[ ]$^{(b,c)}$ < 3 S_m = 80.1 ksi	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$ < 1.0	[ ]$^{(b,c)}$
3. Closure Studs	[ ]$^{(b,c)}$ < 3 S_m = 80.1 ksi	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$ < 1.0	[ ]$^{(b,c)}$
CRDM Housings	[ ]$^{(b,c)}$ < 3 S_m = 69.9 ksi	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$ < 1.0	[ ]$^{(b,c)}$
Vent Pipe	[ ]$^{(b,c)}$ < 3 S_m = 69.9 ksi	N/A	[ ]$^{(b,c)}$ < 1.0	N/A
Bottom Head Juncture	Shell: [ ]$^{(b,c)}$ < 3 S_m = 80.1 ksi Head: [ ]$^{(b,c)}$ < 3 S_m = 80.1 ksi	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$ < 1.0	[ ]$^{(b,c)}$
Bottom Head Instrumentation Tubes	[ ]$^{(b,c)}$ < 3 S_m = 69.9 ksi	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$ < 1.0	[ ]$^{(b,c)}$
Main Vessel Shell	[ ]$^{(b,c)}$ < 3 S_m = 80.1 ksi	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$ < 1.0	[ ]$^{(b,c)}$
Core Support Pads	Shell: [ ]$^{(b,c)}$ < 3 S_m = 80.1 ksi Head: [ ]$^{(b,c)}$ < 3 S_m = 80.1 ksi	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$ < 1.0	[ ]$^{(b,c)}$
Head Adapter Plug	[ ]$^{(b,c)}$ < 3 S_m = 48.3 ksi	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$ < 1.0	[ ]$^{(b,c)}$

Table 5.1.1-2 Faulted Condition Calculations for the Core Support Pads		
Maximum General Primary Membrane Stress Intensity, P_m		**Previous**
Core Support Pad	$P_m = [\quad]^{(b,c)} < 2.4S_m = 55.92$ ksi	N/A
Vessel Shell	$P_m = [\quad]^{(b,c)} < 0.7S_U = 56.0$ ksi	N/A
Maximum Primary Membrane plus Bending Stress Intensity, $P_L + P_b$		**Previous**
Core Support Pad	$P_L + P_b = [\quad]^{(b,c)} < 2.4S_m = 83.88$ ksi	N/A
Vessel Shell	$P_L + P_b = [\quad]^{(b,c)} < 1.05S_U = 84.0$ ksi	N/A

5.1.2 Reactor Vessel Integrity

This analysis is outside the Westinghouse scope of supply on the SGR/Uprating program. This information will be provided by CP&L or other vendors.

5.2 Reactor Internals

5.2.1 Introduction

The reactor pressure vessel (RPV) system consists of the reactor vessel, reactor internals, fuel and control rod drive mechanisms. The reactor internals function to support and orient the reactor core fuel assemblies and control rod assemblies, absorb control rod assembly dynamic loads, and transmit these and other loads to the reactor vessel. The reactor vessel internal components also function to direct coolant flow through the fuel assemblies (core), to provide adequate cooling flow to the various internals structures, and to support in-core instrumentation. They are designed to withstand forces due to structure deadweight, preload of fuel assemblies, control rod assembly dynamic loads, vibratory loads, earthquake accelerations, and Loss of Coolant Accident (LOCA) loads.

Operating a plant at conditions (power, pressure, temperature, flow) other than those considered in the original design requires that the reactor vessel/internals/fuel system interface be addressed in order to assure compatibility and that the structural integrity of the reactor vessel/internals/fuel system is not adversely affected. In addition, thermal-hydraulic analyses are required to determine plant specific core bypass flows, pressure drops and upper head temperatures in order to provide input to the LOCA and non-LOCA safety analyses as well as NSSS performance evaluations.

Generally, the areas of concern most affected by changes in system operating conditions are:

1. Reactor internals system thermal/hydraulic performance;
2. Rod cluster control assembly (RCCA) drop time performance; and
3. Reactor internals system structural response and integrity

The major components and features of the reactor internals system for HNP are summarized as follows. The lower core support assembly consists of the lower support plate, lower support columns, and lower core plate, and supports the fuel assemblies on the sides and at the bottom. The guidance and alignment of the lower core support assembly during insertion into the reactor vessel is provided by the radial support system and the head-vessel alignment pins, and special temporary guide studs attached to the vessel. The hold-down spring rests on top of the flange of the lower core support assembly. The upper core support assembly consists of the upper support plate, upper support columns, and upper core plate, and rests on top of the hold down spring. The guidance and alignment of the upper core support assembly during its insertion is provided by the head-vessel alignment pins, the upper core plate alignment pins in the core barrel assembly, and the special temporary guide studs attached to the vessel. The alignment of the core, i.e., each fuel assembly, is provided through the engagement of the lower core plate fuel pins into the bottom of the fuel assemblies and the upper core plate fuel pins into the top of the fuel assemblies. The vessel upper head compresses the hold-down spring providing joint preload.

The core barrel, which is part of the lower core support assembly, provides a flow boundary for the reactor coolant. When the primary coolant enters the reactor vessel, it impinges on the side of the core barrel and is directed downward through the annulus formed by the gap between the outside diameter of the core barrel and the inside diameter of the vessel. The flow then enters the lower plenum area between the bottom of the lower support plate and the vessel bottom head and is redirected upward through the core. After passing through the core, the coolant enters the upper core support region and then proceeds radially outward through the reactor vessel outlet nozzles. Another portion of the primary coolant bypasses the fuel and cools the upper head region. The perforations in the various components, such as the lower support plate, control and meter the flow through the core.

HNP currently uses Siemens Power Corporation (SPC) fuel. For certain calculations or evaluations, properties of Westinghouse 17 x 17 VANTAGE 5H (V5H) fuel were used in place of the SPC fuel properties. The combination of the Westinghouse and Siemens fuel properties result in a bounding evaluation.

The following sections summarize the work performed to assess the effect on the reactor pressure vessel/internals system due to the SGR/Uprating at HNP.

5.2.2 Description of Analyses and Evaluations

5.2.2.1 Thermal/Hydraulic System Evaluations

5.2.2.1.1 System Pressure Losses

The principal reactor coolant system flow route through the HNP reactor pressure vessel system begins at the three inlet nozzles. At this point, flow turns downward through the reactor vessel core barrel annulus. After passing through this downcomer region, the flow enters the lower reactor vessel dome region. This region is occupied by the internals energy absorber structure, lower support columns, bottom-mounted instrumentation columns, and supporting tie plates. From this region, flow passes upward through the lower core plate, and into the core region. After passing up through the core, the coolant flows into the upper plenum, turns, and exits the reactor vessel through the three outlet nozzles. Note that the upper plenum region is occupied by support columns and RCCA guide columns.

A key area in evaluation of core performance is the determination of hydraulic behavior of coolant flow within the reactor internals system, (i.e., vessel pressure drops, core bypass flows, RPV fluid temperatures and hydraulic lift forces). The pressure loss data is necessary input to the LOCA and non-LOCA safety analyses and to overall NSSS performance calculations. The hydraulic forces are critical in the assessment of the structural integrity of the reactor internals, core clamping loads generated by the internals hold down spring, and the stresses in the reactor vessel closure studs.

The THRIVE computer code (Reference 1) was used to perform this evaluation by solving the mass and energy balances for the HNP reactor internals fluid system. The analysis determined the distribution of pressure and flow within the reactor vessel, internals, and reactor core for the RCS at the SGR/Uprating conditions. The THRIVE computer code has been used for this analysis since the original plant design.

The analysis determined the distribution of pressure and flow within the reactor vessel, internals and the reactor core for the RCS at the SGR/Uprating conditions.

5.2.2.1.2 Bypass Flow Analysis

Bypass flow is the total amount of reactor coolant flow bypassing the core region and is not considered effective in the core heat transfer process. Since variations in the size of some of the bypass flow paths, such as gaps at the outlet nozzles and the core barrel, occur during manufacturing or change due to different fuel assembly designs or due to changes in the RCS conditions, plant specific as-built dimensions are used in order to demonstrate that the bypass flow limits are not violated. Therefore, analyses are performed to determine core bypass flow values to either show that the design bypass flow limit for the plant will not be exceeded or to determine a revised design core bypass flow.

The present design core bypass flow limit is 7.1 percent of the total reactor vessel flow. The purpose of this evaluation is to ensure that the design value of 7.1 percent can be maintained at the uprated RCS conditions. The principal core bypass flow paths are:

- Baffle-Barrel Region

  The HNP reactor vessel internals incorporate an upflow configuration. Reactor coolant flow enters the baffle-barrel region at the bottom former elevation and passes through the flow holes at each successively higher former elevation, past the upper core plate and into the outlet plenum. For the upflow configuration, all the flow that enters the baffle-barrel region is considered core bypass flow.

- Vessel Head Cooling Spray Nozzles

  These nozzles are flow paths between the reactor vessel and core barrel annulus and the fluid volume in the vessel closure head region above the upper support plate. A fraction of the flow that enters the vessel inlet nozzles and into the vessel/barrel downcomer passes through these nozzles and into the vessel closure head region. The purpose of these flow paths is to allow circulation of a small fraction of the cold leg coolant into the upper head region of the reactor vessel.

- Core Barrel - Reactor Vessel Outlet Nozzle Gap

  At HNP some of the flow that enters the vessel/barrel downcomer leaks through the gaps between the core barrel outlet nozzles and the reactor vessel outlet nozzles and merges with the vessel outlet nozzle flow. Since the lower reactor internals are designed to be removable from the reactor vessel, a small circumferential gap exists at each of the outlet nozzle locations. While the gap is designed to be very small and closes down somewhat at operating conditions due to the differential coefficient of thermal expansion between the reactor internals and the reactor vessel, there is some amount of flow which leaks directly from the vessel inlet/downcomer region and out these nozzle gaps.

- Fuel Assembly - Baffle Plate Cavity Gap

  The baffle plates surround the reactor fuel assemblies or core region. The gap between the peripheral fuel assemblies and the baffle plates is referred to as the core cavity region. This is the core bypass flow path between the peripheral fuel assemblies and the core baffle plates.

- Fuel Assembly Thimble Tubes

  Thimble tubes are used as paths for the insertion and removal of control rods, thimble plugging devices and various core components such as burnable absorbers. These tubes are physically part of each fuel assembly and flow within them is partially effective in removing core heat. However, such flow is analytically not considered to be effective in heat removal, and is consequently core bypass flow.

Fuel assembly hydraulic characteristics, system parameters, such as inlet temperature, reactor coolant pressure and flow were used in the calculation to determine the impact of the new SGR/Uprating conditions on the total core bypass flow. The total core bypass flow value (including uncertainties) was determined to be 6.95 percent. Therefore, the design core bypass flow value of 7.1 percent of the total vessel flow is maintained.

5.2.2.1.3 Hydraulic Lift Forces

The reactor internals hold-down spring is essentially a large diameter belleville type spring of rectangular cross section. The purpose of this spring is to maintain a net clamping force between the reactor vessel head flange and upper internals flange and the reactor vessel shell flange and the core barrel flange of the internals. An evaluation was performed to determine hydraulic lift forces on the various reactor internal components to ensure that the reactor internals assembly would remain seated and stable for all conditions. A calculation was performed to determine if the hydraulic lift forces with SPC fuel are comparable to or less than those previously evaluated using Westinghouse 17x17 V5H fuel.

The results of this evaluation indicate that the SPC fuel lift forces are bounded by the Westinghouse V5H forces previously evaluated. Therefore, the HNP reactor internals assembly would remain seated and stable for all conditions under SGR/Uprating conditions. The SGR/Uprating effects on the SPC fuel are further addressed in Section 5.3.

5.2.2.1.4 RCCA Drop Time Performance Evaluation

The RCCAs represent perhaps the most critical interface between the fuel assemblies and the other internals components. It is imperative to show that the new RCS conditions will not adversely impact the operation of the control rods, either during accident conditions or normal operation.

The purpose of this section is to determine the potential impact due to the SGR/Uprating at HNP on RCCA drop time characteristics used in the FSAR for accident analyses. This analysis is based on 17 x 17 V5H fuel assemblies with intermediate flow mixing (IFM) grids.

Calculations were performed which indicated that the maximum drop time-to-dashpot entry of 2.7 seconds for HNP remains conservatively applicable for accident analyses.

5.2.2.1.5 Momentum Flux and Fuel Rod Stability

No detailed treatment of momentum flux and associated fuel rod stability is needed due to the upflow configuration in the baffle-barrel region. There is very minimal pressure drop across the baffle plates since the flow is upward on both sides, and pressure relief holes further reduce the pressure differential across the plates.

5.2.2.2 Mechanical System Evaluations

The evaluation of the mechanical response of the RCS subjected to auxiliary line breaks of a LOCA transient is performed in three steps. First, the RCS is analyzed for the effects of loads induced by normal operation, which includes thermal, pressure, and deadweight effects. From this analysis, the mechanical forces acting on the RPV that would result from the release of the equilibrium forces at the break locations are obtained. In the second step, the loop mechanical loads and the reactor internals hydraulic forces are simultaneously applied, and the RPV displacements due to the LOCA are calculated. Finally, the structural integrity of the reactor coolant loop and component supports to deal with the LOCA is evaluated by applying the reactor vessel displacements to a mathematical model of the reactor coolant loop. Thus, the effects of vessel displacements upon loop and reactor vessel and its internals are evaluated.

Since HNP takes credit for leak-before-break (LBB) applied to the primary loop (Reference 2), the LOCA analyses of the reactor pressure vessel system for postulated ruptures of the primary loop piping are not required. The next limiting breaks to be considered are the branch line breaks which consists of (a) accumulator line, (b) pressurizer surge line, and (c) residual heat removal

(RHR) line breaks. Of these branch line breaks, the most limiting breaks considered for the dynamic analysis of the HNP reactor pressure vessel system are the accumulator line break (cold leg) and the pressurizer surge line break (hot leg).

The analysis model of the RPV is a three-dimensional nonlinear finite element model that represents the dynamic characteristics of the reactor vessel and its internals in the six geometric degrees of freedom. The model was developed using the WECAN computer code (Reference 3). The WECAN computer code has been used for this analysis since the original plant design.

5.2.2.2.1 Loss-of-Coolant Accident (LOCA) Loads

The LOCA loads applied to the HNP reactor pressure vessel system consist of (1) reactor internal hydraulic loads (vertical and horizontal), (2) reactor coolant loop mechanical loads, and (3) pressure loads acting on the baffle plates. All the loads are calculated and combined in a time-history manner. The MULTIFLEX computer code (Reference 4) was used to calculate the hydraulic transients within the entire primary coolant system (see Section 6.6). The MULTIFLEX code has been used for this analysis since the original plant design.

The severity of a postulated break in a reactor vessel is related to two factors: the distance from the reactor vessel to the break location, and the break opening area. The nature of the reactor vessel decompression following a LOCA, as controlled by the internals structural configuration previously discussed, results in larger reactor internal hydraulic forces for pipe breaks in the cold leg than in the hot leg (for breaks of similar area and distance from the RPV). Pipe breaks farther away from the reactor vessel are less severe because the pressure wave attenuates as it propagates toward the reactor vessel. With the consideration of LBB, the auxiliary line breaks, such as the accumulator line break and the pressurizer surge line break, are not as severe as the main line breaks (e.g., RPV inlet nozzle or RCP outlet nozzle break).

The reactor vessel impact forces were calculated. The reactor pressure vessel peak displacements and rotations for the accumulator line break, the pressurizer surge line break, and the RHR line break were also determined. The impact forces calculated at the vessel/internals interfaces were used to evaluate the structural integrity of the reactor vessel and its internals. The core plate motions for this analysis were provided as input to the fuel assembly grid load evaluation. The resultant fuel assembly loads were used in additional fuel evaluations. (See Section 5.3.)

5.2.2.2.2 Flow-Induced Vibrations

Flow-induced vibrations (FIV) of pressurized water reactor internals have been studied at Westinghouse for a number of years. The objective of these studies was to demonstrate the structural integrity and reliability of reactor internal components.

Lower Internals Response

Results from the scale model and in-plant tests indicate that the primary cause of lower internals' excitations is the flow turbulence generated by the expansion and turning of the flow at the transition from the inlet nozzle to the barrel-vessel annulus, and the wall turbulence generated in the downcomer.

The response of the HNP lower internals to the uprated power can possibly be influenced by changes in fuel assembly mechanical characteristics and NSSS parameters. The impact of these factors on the response of the HNP lower internals is discussed in the following sections.

Impact of NSSS Parameters

The NSSS parameters that could potentially influence the FIV response of the reactor internals include the inlet nozzle flow velocities, vessel/core inlet temperatures, and the vessel outlet temperatures.

Generally, the inlet nozzle velocity for the FIV response during hot functional testing is calculated at the mechanical design flows, which are approximately 15 percent higher than the thermal design flows. Since the thermal design flows for the SGR/Uprating are lower than the original thermal design flowrates and since HNP hot functional tests were performed at flowrates higher than those for the SGR/Uprating conditions, the existing test results are conservative and remain applicable.

The other parameter that would influence the FIV response is the core inlet temperature. For the most limiting case of the SGR/Uprating changes, the vessel/core inlet temperature is 536.6°F. The originally analyzed vessel/core inlet temperature is 543.5°F. This temperature change implies a change in water density, which had a negligible impact on the core barrel response due to the SGR/Uprating conditions.

Upper Internals Response

The significant flow-induced forces on upper internals are due to random turbulence generated by the cross flows that converge on the outlet nozzles. Therefore, the guide tubes and the upper support columns that lie in the vicinity of the outlet nozzles will experience the maximum flow-induced forces.

The flow-induced vibration loads on the guide tubes and the upper support columns increase by approximately 2.8 percent for the SGR/Uprating. Previous FIV analyses on the guide tubes and the upper support columns have shown that sufficient margins exist to accommodate this increase in the FIV loads. Consequently, the structural integrity of the HNP reactor internals remains acceptable with regard to flow-induced vibrations.

5.2.2.3 Structural Evaluation of Reactor Internal Components

5.2.2.3.1 Introduction

In addition to supporting the core, a secondary function of the reactor vessel internals assembly is to direct coolant flows within the vessel. While directing the primary flow through the core, the internals assembly also establishes secondary flow paths for cooling the upper regions of the reactor vessel and for cooling the internals structural components. Some of the parameters influencing the mechanical design of the internals lower assembly are the pressure and temperature differentials across its component parts and the flowrate required to remove the heat generated within the structural components due to radiation (e.g., gamma heating). The configuration of the internals provides for adequate cooling capability. Also, the thermal gradients, resulting from gamma heating and core coolant temperature changes, are maintained below acceptable limits within and between the various structural components.

Structural evaluations are required to demonstrate that the structural integrity of the reactor components is not adversely affected directly by the change in RCS conditions and transients and/or by secondary effects of the change on reactor thermal hydraulic or structural performance. The presence of heat generated in reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth, which must be accounted for in the design and analysis of the various components.

Since the HNP reactor internals were designed prior to the introduction of Subsection NG of the ASME Boiler and Pressure Code Section III, a plant-specific stress report on the reactor internals was not required. However, the design of the HNP reactor internals was evaluated according to the Westinghouse internal criteria which were similar to the criteria described in Subsection NG of the ASME Code, 1989 edition, 1990 addenda. Moreover, the structural integrity of the HNPs reactor internals design has been shown by analyses performed on both generic and plant specific bases. These analyses were used as the basis for the evaluation of the critical reactor internal components for the plant uprating and the revised thermal transients.

5.2.2.3.2 Component Evaluations

In addition to the loads developed as a result of the thermal performance of the reactor pressure vessel system and components, the internals will also experience other loadings, for example:

- Pressure differentials due to coolant flow
- Weight of the structure
- Superimposed loads from other components
- Earthquake (or seismic) loads,

- LOCA loads,
- Vibratory loads, and
- Preloads.

The purpose of this section of the report is to summarize the work performed to assess the impact of the SGR/Uprating program on the structural integrity of the critical reactor internal components, such as the lower core plate and the upper core plate.

5.2.2.3.2.1 Lower Core Plate Evaluation

A fatigue evaluation was performed for the lower core plate, which is known to be the most critically cycled component in the reactor internals assembly (i.e., cumulative fatigue usage closest to 1).

Structural evaluations were performed to demonstrate that the structural integrity of the lower core plate is not adversely affected directly by the change in RCS conditions and/or by secondary effects of the change on reactor thermal hydraulic or structural performance. For this lower core plate evaluation, the criteria described in Section III, Subsection NG of the ASME Code, 1989 Edition, 1990 Addenda (Reference 5) were utilized.

The method of analysis of the lower core plate consisted of the following three steps:

1. Determination of temperature distributions in the plate

2. Determination of stresses in the plate

3. Determination of margin of safety and fatigue usage factor for the most severely stressed location of the plate

The ANSYS computer code (Reference 6) was used for the thermal and stress analyses performed on the lower core plate. ANSYS is an industry-accepted code for performing finite element analysis. The code has been verified and is under Westinghouse configuration control for use in this application. This is the first application of ANSYS to HNP. The analysis was previously performed by WECAN.

Load Case Combinations

The normal/upset load case combination used is considered to envelop all other possible load combinations. The normal/upset load combination includes:

1. Fuel assembly and core plate deadweight
2. Hold down spring preload

3. Hydraulic lift forces
4. Operating Basis Earthquake (OBE) seismic acceleration
5. Short term heat generation rates
6. Control Rod Drop

Under faulted conditions, LOCA and safe shutdown earthquake (SSE) loads were also considered without secondary loadings. The faulted condition assumes simultaneous occurrence of the SSE and LOCA (loss of coolant accident or blow down) loads. According to the F-1323.1(b) guidelines described in the ASME Section III, Appendix F (Reference 7), only the primary stresses (primary membrane, and primary membrane plus bending) are needed to be considered for the faulted conditions. Note that the SSE and LOCA loads are statistically independent events. The total faulted stress was determined due to stresses combined using the square root of the sum of the squares (SRSS) method. The square root of the sum of the SSE stress squared and LOCA stress squared were calculated before the stress ranges and stress intensities were calculated.

Method of Evaluation

The approach used for this evaluation was to assume the most severe conditions were applied for all of the design transients. By developing the maximum possible thermal stresses, conservative stress intensity margins of safety and fatigue usage resulted. The lower core plate temperatures were determined using the short term heating rates combined with convection to the reactor coolant. In the stress analysis the thermal stresses were combined with the mechanical stresses produced by the mechanical loads. This solution then provided the boundary conditions and structural temperatures for the detailed model. The stress intensity calculated at the bottom of the fuel pin counterbore (which was the location of peak stress intensity) was used to determine the lower core plate fatigue usage.

Discussion of Results Regarding Lower Core Plate

The overall temperature distribution in the lower core plate was obtained at 100 percent steady state full power for the SGR/Uprating.

The temperature distribution followed the flow hole pattern, with the peak temperatures occurring in the larger ligaments in the plate. The heating rates at the perimeter of the plate were low enough that the outer edge of the plate was cooled close to the inlet temperature of the RCS coolant. The outer edge of the plate provided a radial constraint that restricted the inner flow hole region from expanding freely. Subsequently, stresses at the inner region of the core plate were generated. The largest gradients in temperature occurred at the thickest ligaments between flow holes in the inner region.

The stresses calculated were compared to ASME Code allowables to determine the margin of safety. Margin of safety is defined as follows:

$$\text{Margin of Safety} = \frac{\text{Allowable Stress}}{\text{Calculated Stress}} - 1$$

Therefore, a margin of safety equal to zero means that the actual stress is equal to the code allowable stress. In the ASME Code Section III, Subsection NG (Reference 5) the following four conditions were identified:

1. Normal Conditions (Service Level A)
2. Upset Conditions (Service Level B)
3. Emergency Conditions (Service Level C)
4. Faulted Conditions (Service Level D)

Normal operation (Service Level A) conditions include any condition in the course of system startup, operation in the design power range, hot standby and system shutdown, other than Upset, Emergency, Faulted or Testing conditions. Upset (Service Level B) occurrences include any deviations from Normal conditions anticipated to occur often enough that the design should include a capability to withstand the conditions without operational impairment. Emergency (Service Level C) conditions include those deviations from normal conditions that require shutdown for correction of the condition or repair. Faulted (Service Level D) conditions include those combinations of conditions associated with extremely low probability postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that consideration of the public health and safety are involved.

An actual margin of safety against failure is assured by the inherent conservatism in the code stress limits. The margins of safety for the lower core plate were determined for the normal, upset, emergency and faulted conditions.

The fatigue usage factor for HNP is evaluated by:

$$U_f = \sum_{i=1}^{M} \frac{n_i}{N_i}$$

where M is the total number of stress cycling ranges considered, n_i is the number of cycles allocated for range i, and N_i is the allowable cycles corresponding to the alternating stress for range i. For the present analysis, M=3. The ASME code acceptance criterion is $U_f < 1.0$.

The margins of safety and cumulative fatigue usage factors are shown in Table 5.2-1.

Conclusions Regarding Lower Core Plate

The objective of the lower core plate evaluation was to determine the effects on the structural integrity of the lower core plate due to SGR/Uprating conditions. The conclusion of these evaluations is that the structural integrity of the lower core plate is maintainable. The evaluations of the new reactor coolant system conditions, which are due to the SGR/Uprating conditions, demonstrate acceptable margins of safety and acceptable fatigue utilization factors for all loading conditions.

5.2.2.3.2.2 Upper Core Plate Evaluations

The upper core plate positions the upper ends of the fuel assemblies and the lower ends of the control rod guide tubes, thus serving as the transitioning member for the control rods in entry and retraction from the fuel assemblies. It also controls coolant flow in its exit from the fuel assemblies and serves as a boundary between the core and the exit plenum. The upper core plate is restrained from vertical movement by the upper support columns that are attached to the upper support plate assembly. The lateral movement is restrained by four equally spaced core plate alignment pins.

The stresses in the upper core plate are mainly due to hydraulic, seismic, and thermal loads. The total thermal stresses are due to thermal bending moments through the thickness and surface peak stresses. Evaluations were performed to determine the impact of the uprating program on the structural integrity of the upper core plate. As a result of this evaluation, it is concluded that the upper core plate is structurally adequate for the new RCS conditions with regard to the SGR/Uprating program.

5.2.2.3.2.3 Additional Reactor Internal Components

Additional components were not further analyzed in detail because either (a) experience from similar programs and modifications indicate that these components are less critically affected by an uprating; or (b) HNP uses an upflow configuration in the core barrel/baffle region, which results in very minimal pressure differential across the baffle plates and, correspondingly lower baffle-barrel bolt stresses.

A series of assessments for similar programs have been performed for reactor internal components that are not significantly impacted by the SGR/Uprating conditions (and the resulting internal heat generation rates), but are affected by the new RCS conditions due to the primary loop transients. These components are:

- Lower core support plate
- Lower support columns
- Core barrel outlet nozzle

- Core barrel
- Core barrel flange
- Core baffle plate
- Lower radial restraints (clevis inserts)
- Baffle/barrel bolts
- Upper core plate alignment pin
- Upper support columns
- Upper support plate
- Guide tubes and support pins
- Neutron pads

These components are structurally adequate for the new RCS conditions at HNP.

5.2.3 Acceptance Criteria

The applicable criteria for the reactor internals evaluation are:

- Control Rod Drop Analysis: The rod drop time values generated consistent with plant operating parameters and configuration should be within the Tech Spec Limit.

- Flow-Induced Vibration Response: The flow-induced vibration response of reactor internal components, in general, depends upon reactor vessel inlet flow rates (such as mechanical design flow), reactor vessel inlet temperature, reactor vessel outlet temperature, and the fuel assembly design. The response of lower internals (core barrel) depends on the vessel inlet temperature and the inlet flow rates, and the response of upper internals (guide tubes and upper support columns) depends on the vessel outlet temperature and the flow exiting through the outlet nozzles. The acceptance criteria for the flow-induced vibration response is that the stresses from the FIV amplitudes remain within the endurance limit of the material for high cycle fatigue and component loads.

- Fuel Grid Forces: The grid forces were generated, verified, and documented. Analytical results were used in qualifying the fuel. (See Section 5.3.)

- Structural Adequacy of Reactor Internal Components: Subsection NG of the ASME Boiler and Pressure Vessel Code edition applicable to the HNP reactor internals did not include design criteria for the internals. The internals were originally designed to meet the intent of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code with addenda through the Winter 1971 Addenda (Reference 8).

5.2.4 Results and Conclusions

Analyses have been performed to evaluate the effect of the SGR/Uprating on the reactor pressure vessel system and reactor internal components. Specifically, the following changes were assessed:

- SGR/Uprating Steady State RCS Conditions
- SGR/Uprating Internal Heat Generation Rates
- SGR/Uprating Reactor Coolant System Design Transients
- Plant Uprating to a NSSS Core Power Level of 2900 MWt

The calculated total reactor vessel pressure drop values and other pressure loss data were transmitted to interface groups for use in their respective analyses (e.g., LOCA and non-LOCA safety analyses — see Sections 6.1 and 6.2).

The total core bypass flow values (with uncertainties) were determined to be 6.95 percent. Therefore, the design core bypass flow value of 7.1 percent of the total vessel flow can be maintained.

The results of the Hydraulic Lift Forces evaluation indicated that the SPC fuel lift forces are bounded by the Westinghouse V5H forces previously evaluated. Therefore, the HNP reactor internals assembly would remain seated and stable for all conditions under SGR/Uprating. The SGR/Uprating effects on the SPC fuel are further addressed in Section 5.3.

The RCCA performance evaluation indicated that the current 2.7 second RCCA drop time to dashpot entry limit (from gripper release of the drive rod) is satisfied at SGR/Uprating conditions.

Evaluations indicated the SGR/Uprating RCS conditions will not adversely impact the response of reactor internals systems and components due to LOCA excitations and flow-induced vibrations. Reactor internals interface loads and core plate motions were re-calculated and transmitted to the appropriate interface groups for their respective evaluations.

Evaluations of the critical reactor internal components (lower and upper core plates) indicated that the structural integrity of the reactor internals is maintained with the SGR/Uprating RCS conditions.

The results for the SGR/Uprating are consistent with and continue to comply with the current HNP licensing basis/acceptance requirements. The results obtained with the Δ75 replacement steam generators at the uprated NSSS power of 2912.4 MWt bound operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

The critical LOCA loads on reactor internals are caused by LOCA forces input from Section 6.6.5, for which current 2787.4 MWt NSSS power operation is bounded.

5.2.5 References

1. Westinghouse letter MED-RPV-5029, "THRIVE Version - .08 Users' Manual," March 30, 1994. (Westinghouse Proprietary)

2. WCAP-14549, "Technical Justification for Eliminating Large Loop Pipe Rupture as the Structural Design Basis for the Shearon Harris Unit 1 Nuclear Power Plant," December 1996.

3. "WECAN/Plus Users' Manual," Second Edition, December 1, 1992. (Westinghouse Proprietary)

4. WCAP-8708-P/A, "MULTIFLEX, a FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic Structure System Dynamics," September 1977. (Westinghouse Proprietary)

5. ASME Code, Section III, Subsection NG, 1989 Edition, 1990 Addenda.

6. ANSYS Computer Code, Version 5.5.1.

7. ASME Code, Section III, Division 1, Appendices, 1989 Edition.

8. ASME Boiler and Pressure Vessel Code, Section III, 1971 Version including Addenda through the Winter 1971.

Table 5.2-1a Margin of Safety (MS) Values for HNP Lower Core Plate ($MS = S_{allowable}/S_{actual} - 1$)				
	Section A	**Section B**	**Section C**	**Section D**
Allowable Stress, psi	4.86E+04	4.86E+04	4.86E+04	4.86E+04
Previous Maximum Stress,(1) psi	[ ](b,c)	[ ](b,c)	[ ](b,c)	[ ](b,c)
Previous MS (1)	[ ](b,c)	[ ](b,c)	[ ](b,c)	[ ](b,c)
SGR/Uprating Maximum Stress	[ ](b,c)	[ ](b,c)	[ ](b,c)	[ ](b,c)
SGR/Uprating MS	[ ](b,c)	[ ](b,c)	[ ](b,c)	[ ](b,c)

Table 5.2-1b Limiting Fatigue Stress HNP SGR/Uprating (Usage <1.0 is Acceptable)			
Max Alternating Stress,(1) ksi (Reference 8)	**Fatigue Usage (Reference 8)**	**HNP SGR/Uprating Max Alternating Stress, ksi**	**HNP SGR/Uprating Fatigue Usage**
[ ](b,c)	[ ](b,c)	[ ](b,c)	[ ](b,c)

Note:

(1) Sections "A" through "D" are critical ligaments in the lower core plate.

5.3 Fuel Assemblies

This analysis is outside the Westinghouse scope of supply on the SGR/Uprating program. This information will be provided by CP&L or other vendors.

5.4 Control Rod Drive Mechanisms

5.4.1 Introduction

This section addresses the ASME Code structural considerations for the pressure boundary components of the Westinghouse full-length L-106A-1 Control Rod Drive Mechanisms (CRDMs) and seismic sleeves for the Harris Nuclear Plant (HNP) SGR/Uprating conditions.

5.4.2 Description of Evaluations

The Westinghouse full-length L-106A-1 CRDMs were designed and analyzed to meet the Harris specifications (References 1, 2 and 3) and the ASME Code (Reference 4). The SGR/Uprating seismic and LOCA loads remain bounded by the current operating conditions.

The HNP CRDMs are exposed to the RCS cold-leg temperature defined by the vessel/core inlet temperature. The SGR/Uprating parameters that were evaluated indicate that none of the temperatures exceed the original operational basis temperature for the CRDMs. The revised parameters also indicate the RCS pressure remains the same at 2250 psi for all parameter cases. The original analysis for the CRDM was performed for a pressure of 2250 psig. Therefore, the SGR/Uprating NSSS parameters are bounded by the original site specific analyses that are documented in the applicable HNP stress, pressure boundary, and seismic sleeve summary reports.

The NSSS design transients were evaluated for any impact on the site-specific and generic CRDM analyses. The HNP CRDM evaluation of the generic and site-specific reports showed that the fatigue analysis based on the current normal, upset and test transients was performed using the Code (NB-3222.4(d)) waiver of fatigue requirements. The present HNP transients and the SGR/Uprating transients were compared for the normal, upset, emergency, faulted and test conditions.

The comparison showed that the normal and upset conditions contain similar groups of transients, but that some of the specific transients have been modified in number of cycles, pressure range and temperature range. For cases that changed, the effect on the various HNP-specific and generic stress reports was evaluated. Since the generic reports use the Code fatigue waiver, only a significant fluctuation in fatigue is of interest.

The SGR/Uprating transients added the RCS Cold Overpressurization transient. The review of the site-specific stress analysis report for the CRDM lower joint canopy indicated that it becomes a transient with a significant normal operation pressure fluctuation. Therefore, an actual fatigue evaluation was performed for this transient.

Also, review of the generic and HNP-specific CRDM stress and thermal analysis reports revealed that the Unit Loading/Unloading at 5 percent of Full-Power transient resulted in a normal

temperature fluctuation that is considered as significant in the fatigue waiver analysis. Consequently, an additional investigation was required for the SGR/Uprating transients to show that the CRDMs are satisfactory. The evaluation was performed and the CRDMs were found to be acceptable.

It should be noted that the only change to the Emergency and Faulted Conditions is an increase in Steam Generator Tube Rupture from 1 to 6 occurrences. Since, per the ASME B&PV Code (Reference 4), only the stresses and not the cycles are included in the fatigue evaluation, the SGTR transient increase in occurrences has no effect on the current satisfactory evaluation of the CRDMs.

5.4.3 Results

The review of the effects of the SGR/Uprating NSSS parameters and NSSS design transients on the CRDM site specific and generic reports, as described in the previous section, shows that the ASME Code of record (Reference 4) is still met. The E-Spec criteria for the CRDMs per References 1, 2 and 3 remain satisfied.

5.4.4 Conclusions

The results for the SGR/Uprating are consistent with and continue to comply with the current HNP licensing basis/acceptance requirements. These evaluations demonstrate that the CRDMs satisfy the appropriate editions of the ASME Code, stress limits, and fatigue usage criteria. The results obtained with the Δ75 replacement steam generators at the uprated power of 2912.4 MWt bound operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

5.4.5 References

1. Plant Design Specification 955578, Revision 0 (EM 5859), Project: Carolina Power and Light Company, Shearon Harris No. 1 and 2, Equipment: Control Rod Drive Mechanism Model L-106A-1, Westinghouse Nuclear Energy System, Pittsburgh, PA, December 7, 1982.

2. Plant Design Specification 955578, Revision 1 (EM 6001), Project: Carolina Power and Light Company, Shearon Harris No. 1 and 2, Equipment: Control Rod Drive Mechanism Model L-106A-1, Westinghouse Nuclear Energy System, Pittsburgh, PA, April 17, 1984.

3. General Equipment Specification G-677470, Revision 5, Westinghouse Electric Corporation - Nuclear Energy System Design, Pittsburgh, PA, May 14, 1982.

4. "ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components," American Society of Mechanical Engineers, NY, 1974 Edition with Addenda through Summer 1974.

Table 5.4-1 Summary of Fatigue Usage of CRDM Areas	
Area	**Fatigue Usage**
CRDM Generic Report, Lower Joint Canopy Area	[ ]$^{(b,c)}$
Lower Joint Canopy Area, ASN 23 Outside Surface	[ ]$^{(b,c)}$
Lower Joint Head Adapter, ASN 8 Inside Surface	[ ]$^{(b,c)}$
Lower Joint Latch Housing, ASN 14 Inside Surface	[ ]$^{(b,c)}$
Middle Joint Latch Housing, ASN 43 Outside Surface	[ ]$^{(b,c)}$
Middle Joint Rod Travel Housing, ASN 53 Inside Surface	[ ]$^{(b,c)}$
Middle Joint Canopy Area, ASN 69 Inside Surface	[ ]$^{(b,c)}$
Upper Joint Rod Travel Housing, ASN 95 Inside Surface	[ ]$^{(b,c)}$
Upper Joint CRDM Cap, ASN 131 Outside Surface	[ ]$^{(b,c)}$
Upper Joint Vent Plug, ASN 142 Inside Surface	[ ]$^{(b,c)}$
Upper Joint Canopy Area, ASN 118 Inside Surface	[ ]$^{(b,c)}$
CLH Latch Housing, ASN 47 Outside Surface	[ ]$^{(b,c)}$
CLH Cap, ASN 152 Inside Surface	[ ]$^{(b,c)}$
CLH Vent Plug, ASN 142 Inside Surface	[ ]$^{(b,c)}$
CLH Canopy Area, ASN 70 Inside Surface	[ ]$^{(b,c)}$

5.5 Reactor Coolant Loop Piping/Supports and Class 1 Auxiliary Line Piping

The Reactor Coolant Loop Piping and Supports for the SGR/Uprating were evaluated in three areas: the primary piping, equipment nozzles, and supports (Section 5.5.1); the application of leak before break (LBB) criteria to the reactor coolant loop piping (Section 5.5.2); and the Class 1 auxiliary line piping (Section 5.5.3). A discussion of each of these evaluations is provided below.

5.5.1 Reactor Coolant Loop Piping and Supports

5.5.1.1 Introduction

Evaluations have been made for the Harris Nuclear Plant (HNP) reactor coolant loop (RCL) piping and support system to consider the impact of the design parameters (Section 2) for the SGR/Uprating program, including the revised weight and center of gravity (CG) of the replacement steam generators (RSGs). The revised NSSS parameters for the SGR/Uprating were also considered in the RCL piping and support system qualification.

RCL Piping

Conservative Model Δ75 steam generator weights were used in the RCL piping system evaluation. In addition, the revised steam generator upper support stiffness values were used. These revised stiffness values are due to modification between the new steam generator feedwater nozzle and the upper support ring.

During RSG installation, the main feedwater nozzle on the Model Δ75 RSG and the feedwater piping is relocated so that it attaches higher on the RSG than in the original analysis. The effect of the nozzle break at the higher location and the subsequent compartment pressurization and jet impingement effect were also included in the evaluation in Reference 5.

Support System

The objective of the primary equipment/piping support system is to transfer all loads from the RCL equipment and piping to the supporting building structure. These support/restraint-modeling considerations require that the support bearing surfaces be in direct contact with the support component and that the restraints maintain their specified gap in the final hot operating position, normally considered to be 100 percent power.

The computation of stresses and/or loads in the RCL supports for all loading conditions is presented, along with the allowable stresses. The supports considered include the following: steam generator columns and upper and lower lateral, reactor coolant pump columns and tie rods, reactor vessel, and pressurizer. As-built deviations such as material substitutions, misalignments, pipe support attachments, etc., have been incorporated into this evaluation.

5.5.1.2 Description of Analyses and Evaluations

The SGR/Uprating program and its associated parameters were reviewed for impact on the existing design basis. The analyses, methods and criteria used in the existing design basis for HNP continue to be used.

5.5.1.2.1 Thermal Expansion

The SGR/Uprating NSSS design transients were compared with the design transients used for the T-hot Reduction/RSG program. The results of the comparison showed that there are no significant differences between the set of design transients. Also, there is no significant change in the reactor vessel inlet and outlet operating temperatures for the SGR/Uprating program and the temperatures used in the thermal analyses. Therefore, the thermal analyses remain valid.

The existing inadvertent startup and loss of power transient temperatures identified in Section 3.1.2 of Reference 1 are still the governing over-temperature conditions. Effects on the fatigue evaluation are discussed in Section 5.5.1.3.4.

5.5.1.2.2 Design Pressure and Weight

The design pressure, which is also the pressure used in the qualification of the RCL piping, is 2485 psig. Uprating pressure variations around the RCL are all less than those used in the piping qualifications. Since the layout of the RCL piping has not changed from the original plant configuration, except for small attached lines, the distributed weight of the pipe, insulation and fluid remain essentially the same. The new weights of the steam generators and water content are considered to be fully supported by the steam generator columns and have negligible effect on the RCL piping.

5.5.1.2.3 Seismic Analysis

The seismic analysis methodology is discussed in detail in Section 4.2 of Reference 1. Seismic analysis is performed by the time history method to consider effects of the RSGs and the revised weight and center of gravity.

5.5.1.2.4 LOCA and Application of Leak Before Break (LBB)

Eleven LOCA break locations were identified in Table 3-1 of Reference 1. With the application of LBB technology to HNP, only three auxiliary line RCL nozzle break locations remain. They are the Residual Heat Removal line RCL connection, the Safety Injection System Accumulator line RCL connection, and the Pressurizer Surge line RCL connection. Time history RCL LOCA analyses were performed (Section 5.5.1.3.3) for the SGR/Uprating program by applying the LOCA Hydraulic Forcing Functions and the RPV motion developed for the project.

5.5.1.2.5 Loading Combinations

Loading combinations and stress evaluation criteria for the RCL piping are defined in Table 3-2 of Reference 1.

The types of loadings considered for the primary equipment supports include deadweight, thermal expansion, operating pressure, seismic events, LOCA, pipe ruptures and jet impingement loadings. Table 5.5.1-1 presents a summary of the loading combinations and stress limits for the primary equipment support loading conditions.

5.5.1.3 Description of Analyses Areas

5.5.1.3.1 Static Analysis

The static analytical model developed for the previous analyses in Reference 1 was revised to reflect the reactor coolant pump (RCP) column tilt effect and individual SG and RCP column stiffness values with the applicable support configurations. Only these column supports are active during normal operating conditions.

The temperatures consistent with the range identified for the SGR/Uprating program were considered in the thermal expansion analyses, as explained in Section 5.5.1.2.1. Active lateral equipment supports and the revised steam generator upper support stiffness values (due to potential interference between the new steam generator feedwater nozzle and the upper support ring) were also considered. The pressure and deadweight loadings have not changed since the original plant design analysis. The weight of the RSGs is considered to be supported by the steam generator columns for deadweight.

The WESTDYN computer code has been used for this analysis since the original plant design (FSAR Section 3.9.1.2.1 Item a, Section 3.9.1.4.3 and Reference 2).

5.5.1.3.2 Seismic Analysis

The dynamic four-massed steam generator model was developed based on the method described in Section 4.2.1 of Reference 1 to reflect the RSG with the revised CG and weight. A dynamic time history seismic analysis for the RSG was performed.

The WECAN computer code (Reference 3) has been used for the seismic analysis since the original plant design (FSAR Section 3.9.1.2.1 Item f).

5.5.1.3.3 LOCA Analysis and Pipe Thrust Force

Time history RCL LOCA analyses were performed for the SGR/Uprating program by applying the LOCA auxiliary line branch nozzle break Hydraulic Forcing Functions and the RPV motions.

The WESTDYN computer code used for this analysis has been used for this analysis since the original plant design (FSAR Section 3.9.1.2.1 Item a and Section 3.9.1.4.3). Broken and Unbroken Loop time history LOCA analyses were performed. With the application of LBB technology, the main loop breaks are eliminated.

For the replacement steam generator (RSG), the feedwater nozzle is located higher on the shell than for the original steam generator. It is located close to the steam generator upper support elevation, and a postulated nozzle break will cause the reactor coolant system to respond differently. Changes to steam generator support loads and changes to displacements of the RSG and attached piping are also expected. In the original plant configuration, the feedwater nozzle was located much lower on the steam generator shell, and the response of the system was much different. Consequently, a new RCL system analysis for the feedwater nozzle break was performed. Since the steam generator compartment walls are closer to the steam generator shell at the new feedwater nozzle elevation, it was necessary to consider compartment pressures and their effects on the steam generator. Compartment differential pressure forces across the steam generator shell were determined and combined with all other forces occurring from the break. The applied loads considered were thrust forces from the fluid escaping from the steam generator, jet impingement forces onto the steam generator from the broken side of the whipped pipe, and differential compartment forces around the steam generator. The analysis method performed was a dynamic time history analysis, with forces applied onto the steam generator model at various elevations and thrust/jet forces applied at the feedwater nozzle with respect to time. Results from this feedwater nozzle break analysis show that the reactor coolant piping is qualified. In addition, all equipment nozzles, displacements at all Class 1 auxiliary piping attached to the RCL, and equipment support loads were shown to be acceptable.

Feedwater break embedment loads and compartment pressure loads were provided for evaluation of compartment walls and structures. Steam generator displacements were also provided for evaluation of piping attached to the RSG. See the BOP Licensing Report.

No new main steamline break analysis was performed for the SGR/Uprating program, as there is no significant change in either the frequencies or the mass.

5.5.1.3.4 Fatigue Analysis

Since the temperature and pressure changes for the SGR/Uprating NSSS design transients are small, changes to the forces and moments for normal operating thermal expansion and operating pressure are also small, and are judged to be insignificant. The forces and moments for deadweight remain the same because the RCL piping properties have not changed. The change in seismic loads from the analysis due to the change in RSG weight and CG to the existing Model D-4 steam generator analysis is also insignificant. Furthermore, the comparison of the NSSS design transients for the SGR/Uprating with the design transients used for the T-hot Reduction/RSG program showed that there is no change in the number of cycles for the design transients. Therefore, the fatigue evaluations for the RSG done for the T-hot Reduction program

remain valid and the fatigue usage factors for all selected locations are below the allowable. The results for the RCL piping stress analysis are summarized in Table 5.5.1-2.

The WESTDYN computer code has been used for the fatigue analysis since the original plant design (FSAR Section 3.9.1.2.1 Item a and Section 3.9.1.4.3).

5.5.1.3.5 Support Analysis

This section presents the structural evaluation of the as-built primary equipment supports for the SGR/Uprating design conditions. The evaluation of the primary equipment supports compares the stresses associated with loads obtained from piping system analyses with the acceptance criteria set forth by the 1974 ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, including all Addenda up to and including Winter 1974 (Reference 4). In addition, for finding material strengths, "Cases of ASME Boiler and Pressure Vessel Code," Case 1644-4, approved by Council, March 1, 1976 was used (Reference 5).

The static and dynamic structural support analyses assume linear elastic behavior; they employ the matrix method and normal mode theory for the solution of lumped-parameter multi-mass structural models. The complexity of the physical system to be analyzed requires the use of a computer for solution. The equipment support structure models are dual-purpose since they are required to: (1) quantitatively represent, in terms of spring stiffness matrices, the elastic restraints which the supports impose upon the loop and (2) evaluate the individual support member stresses due to the forces imposed on the supports by the loop.

To accurately represent the elastic behavior of the supports, structural geometry, topology, member releases, and concrete flexibility are all considered in developing the support stiffness characteristics. These support stiffness characteristics are included in the RCL model.

To evaluate the effect of the support loads obtained from the RCL analysis, each element of the support that would assist in transferring these loads to the building structure concrete was considered.

The steam generator lower lateral supports are a series of bumpers that act in compression. Each was designed and evaluated as a linear-type support according to the ASME Code (Reference 4). Elements of this system include the struts that transfer the loads directly to the building structure concrete and the cross-compartment beam. To ensure the integrity of the beam, the built-up steel section, the beam splice bolts, and the anchorage were all considered in the evaluation.

The steam generator upper supports consist of a main circular ring band that transfers loadings at this elevation to a series of bumpers and hydraulic snubbers. Each was evaluated as a linear-type support that is active during seismic and LOCA events. Capacities of each element represent the capacity of the support system in the three orthogonal directions. The back-up structures for the

SG upper lateral support ring band/snubbers assemblies are supplied by others and are not included in the evaluation.

The steam generator and reactor coolant pump columns are support members that transfer tensile or compressive loads from the equipment to the building structure concrete. These supports were designed and evaluated as linear-type supports as defined in the ASME Code (Reference 4). The support elements considered were the Type A and B forgings, the shear pins, column clevis plates, the pipe column section and the column base assembly. Tensile and compressive capacities of each element were generated and compared to one another to determine the critical load capacity of this support.

The reactor coolant pump is supported laterally by the tie rods. These tie rods carry pump lateral loads in the form of tensile loads to the building structure concrete and were considered as linear-type supports. Capacities of the Type A column adapter forgings, the tie rods, shear pins, and the tie rod brackets were developed and compared to determine the critical load-carrying element.

The cross-compartment beams are modeled as linear elastic supports for analysis with the STRUDL computer code (Reference 6). The STRUDL computer code has been used for this analysis since the original plant design (HNP FSAR subsection 3.9.1.4.4).

The reactor vessel support system consists of six identical support members. Each member has three elements that were evaluated. The shoe and the shear pins were evaluated as linear-type supports with the box structure evaluated using the plate and shell criteria of the ASME Code (Reference 4). The finite element analysis of the box structure, produced primary membrane (P_m) and primary membrane plus bending ($P_m + P_b$) stresses for each finite element of the model.

Vertical and horizontal loads are transmitted from the reactor vessel nozzle pad to the reactor vessel support box structure. Each box structure consists of a horizontal top plate, which receives loads from the forged reactor vessel shoe. Intermediate vertical plates transmit these loads to a horizontal bottom plate. The box was modeled as a plate-and-shell-type support for a finite element analysis with the WECAN computer code (Reference 3). The WECAN computer code has been used for this analysis since the original plant design (HNP FSAR subsection 3.9.1.4.4).

5.5.1.4 Acceptance Criteria

The design criteria used for the piping qualification of the HNP RCL system are defined in the design specifications. They are based on the ASME B&PV Code Section III (NB) through the Winter 1979 Addenda (Reference 7).

Table 5.5.1-1 presents a summary of the loading combinations and stress limits for the primary equipment support evaluation.

5.5.1.5 Results

5.5.1.5.1 Piping Stresses and Fatigue Qualification

The deadweight and thermal analysis results for the RCL piping are summarized in Table 5.5.1-2.

Differences between the displacements from the seismic analyses due to the RSG (with the revised weight and center of gravity) and the displacements from the existing Model D4 steam generator analysis are insignificant. The evaluation showed that the Model D4 existing steam generator seismic analysis support loads envelop the seismic support loads for the RSG.

The fatigue analysis results remain valid, as discussed in Section 5.5.1.3.4.

The stresses due to primary loadings of pressure, weight, OBE, SSE, and LOCA are combined in accordance with the methods described in Section 5.1 of Reference 1. The maximum primary stresses due to pressure, weight, OBE, SSE, and LOCA in the RCL piping and the ASME Code allowable stress values are presented in Table 5.5.1-2. The table also provides the maximum secondary or fatigue stresses, the usage factors, and the ASME Code allowable values.

The RCL piping stresses are all acceptable.

5.5.1.5.2 Primary Equipment Nozzle Loads

The primary equipment nozzle loads are compared to the allowable nozzle loads provided in the equipment design specifications. In cases where the nozzle loads exceed the allowable values, principal stresses are calculated to qualify the primary equipment nozzle loads. The primary equipment nozzle loads are all acceptable.

5.5.1.5.3 Leak Before Break (LBB)

The LBB loads are acceptable as indicated by Reference 8. The LBB loads qualification is presented in Section 5.5.2 of this report.

5.5.1.5.4 As-built Condition and Pipe Wall Thickness Issues

The existing as-built condition has been reconciled in Section 5.4 of Reference 1. The pipe wall thickness issues have been considered in the analyses and found to be acceptable. The results are incorporated in Table 5.5.1-2.

5.5.1.5.5 Primary Equipment Supports

Table 5.5.1-3 defines the maximum support stresses for the steam generator columns, lower and upper lateral, reactor coolant pump columns and tie rods, reactor vessel support, and the

pressurizer supports. Member loads are shown for each support with its corresponding maximum stress level expressed as a percentage of the allowable stress. For the faulted loading condition, the controlling pipe rupture or LOCA break case is listed.

The stress evaluation for the Upset conditions shows that the stresses are below the allowable values established by the ASME Code (Table 5.5.1-3). The contribution to the Faulted condition stresses is predominantly from the LOCA and seismic loadings; the contribution from the loadings due to pressure and weight is relatively minor. The supports system stress evaluation for the Faulted condition shows that the stresses are below the allowable values for all support members as established in the Code.

5.5.1.6 Conclusions

A piping stress analysis of the RCL was performed for HNP to consider the design parameters of the SGR/Uprating program, and the design has been found to be adequate. The RCL system will maintain its structural integrity and meet all safety related design requirements.

The analysis results show that the primary stresses are below the allowable limits established in the Code (Reference 7) under the design, upset, and faulted conditions. The RCL piping secondary stresses and fatigue usage factors are in conformance with the requirements of the Code for the fatigue damage evaluation performed under all normal, upset, and test conditions. The primary equipment nozzle loads are all acceptable. Therefore, the RCL piping system is adequate for all design loading conditions.

The as-built primary equipment supports system is adequate. It is calculated to maintain its structural integrity and meet the safety-related design requirements under all specified design conditions, as defined by the ASME Code (Reference 4), the original code of record for the HNP primary equipment supports. The stress report was updated in Reference 9.

Operation of HNP with the Δ75 replacement steam generators at the uprated power of 2912.4 MWt bounds operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt. Operation at the uprated NSSS power is bounding since the associated NSSS parameters (e.g., maximum and minimum temperatures) and the design transients of concern for this evaluation result in maximum stresses that remain acceptable to the ASME criteria and would bound those determined at the lower power level.

5.5.1.7 References

1. WCAP-9990, Volume 1, Rev. 2 "Structural Analysis of the Reactor Coolant Loop for Shearon Harris Nuclear Power Plant Unit 1 - Analysis of the Reactor Coolant Loop Piping," September 1988.

2. Westinghouse Letter EDRE-SMT-98-121, "Release of WESTDYN Version 7.1," W. R. Morrison, November 3, 1998.

3. WECAN - Westinghouse Electric Computer Analysis, Version Date 1997.

4. ASME Boiler and Pressure Vessel Code, Section III (NF), including Addenda up to and including Winter 1974, "Nuclear Power Plant Components," American Society of Mechanical Engineers, New York, 1974.

5. "Cases of ASME Boiler and Pressure Vessel Code," Case 1644-4, approved by Council March 1, 1976.

6. User Manual for ICES-STRUDL-II, Version 2-M2, June 1972.

7. ASME Boiler and Pressure Vessel Code, Section III (NB), Through Winter of 1979 Addenda.

8. WCAP-14549, "Technical Justification For Eliminating Large Primary Loop Pipe Rupture As The Structural Design Basis For The Shearon Harris Unit 1 Nuclear Power Plant," December 1996.

9. WCAP-9990, Volume 2, Rev. 1, "Structural Analysis of the Reactor Coolant Loop for Shearon Harris Nuclear Station Units No. 1 and 2," March 2000.

Table 5.5.1-1

Support Structure Loading Combinations and Stress Criteria For Normal, Upset, and Faulted Conditions

Operating Condition	Loading Combination	Stress Limit	
		Linear-Type Supports	Plate and Shell Supports
Normal	Thermal expansion Weight Operating pressure	Within working limits	$P_m \leq 1.0\ S_m$ $P_m + P_b \leq 1.5\ S_m$
Upset	Thermal expansion Weight Operating pressure OBE Pipe support attachments	Within working limits	$P_m \leq 1.0\ S_m$ $P_m + P_b \leq 1.5\ S_m$
Faulted[3]	Operating pressure Weight LOCA SSE Jet impingement Pipe support attachments	Smaller factor of 2.0 or $1.167\ S_u/S_y$ if $S_u \geq 1.2\ S_y$ or 1.4 if $S_u < 1.2\ S_y$ Times working limits	$P_m \leq 1.5\ S_m$ or $1.2\ S_y$ (greater)[1] $P_m + P_b \leq 2.25\ S_m$ or $1.8\ S_y$ (greater)[2]

Notes:

(1) Not to exceed $0.7\ S_u$

(2) Not to exceed $1.05\ S_u$

(3) Faulted = Deadweight + Pressure $\pm\sqrt{(SSE)^2 + (LOCA)^2}$

Table 5.5.1-2 Piping Stress Analysis Summary						
	Hot Leg		Crossover Leg		Cold Leg	
Evaluation	Maximum	Allowable	Maximum	Allowable	Maximum	Allowable
Equation 9, Design Stress (ksi) (DW, P)	[ ]$^{(b,c)}$	26.32	[ ]$^{(b,c)}$	26.32	[ ]$^{(b,c)}$	26.32
Equation 9, Upset Stress (ksi) (DW, P, OBE)	[ ]$^{(b,c)}$	29.25	[ ]$^{(b,c)}$	29.25	[ ]$^{(b,c)}$	29.25
Equation 9, Faulted Stress (ksi) (DW, P, SSE, LOCA)	[ ]$^{(b,c)}$	52.64	[ ]$^{(b,c)}$	52.64	[ ]$^{(b,c)}$	52.64
Equation 12, Stress (ksi)	[ ]$^{(b,c)}$	53.10	[ ]$^{(b,c)}$	53.10	[ ]$^{(b,c)}$	53.10
Equation 13, Stress (ksi)	[ ]$^{(b,c)}$	53.10	[ ]$^{(b,c)}$	53.10	[ ]$^{(b,c)}$	53.10
Fatigue Usage Factor	[ ]$^{(b,c)}$	1.0	[ ]$^{(b,c)}$	1.0	[ ]$^{(b,c)}$	1.0

Note: All Allowable Stresses in the table are based on design temperature.

Table 5.5.1-3 Primary Equipment Support Member Stresses					
	Member Stresses (Percent of Allowable/Loading Condition) Units (kips)				
	Upset		Faulted		
Member	Load	Percent Stressed	Load	Break	Percent Stressed
SG Lower Support Bumpers	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
SG Lower Support Beams (1, 2)	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
SG Columns (3)	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$
SG Upper Supports (4)	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
RCP Tie Rods	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
RCP Columns (3)	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$
RV Support	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[$^{(5)}$]$^{(b,c)}$	[ ]$^{(b,c)}$
Pressurizer Skirt	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Pressurizer Struts	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$ [ ]$^{(b,c)}$	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$

Notes:

(1) Ratio includes seismic loads, anchor wall motion and restrained thermal expansion (+50°F) of SGLL Beam.

(2) Effects of pipe support attachment loads are included.

(3) (+) = Tension, (–) = Compression for column loadings.

(4) Loads shown are the controlling total resultant loads applied at the centerline of the steam generator

(5) RV support LOCA loads and stresses are conservative

5.5.2 Reactor Coolant Loop Piping LBB Analysis

5.5.2.1 Introduction

The original structural design basis of the Reactor Coolant System (RCS) for HNP required that the dynamic effects resulting from pipe break be considered and that protective measures for such breaks be incorporated into the design. Subsequent to the original HNP design, additional concern regarding asymmetric blowdown loads was raised as described in Unresolved Safety Issue A-2 (Asymmetric Blowdown Loads on the Reactor Coolant System). The resolution of the Unresolved Safety Issue A-2 was identified in Generic Letter 84-04 (Reference 1). However, research by the NRC and industry coupled with operating experience determined that safety could be negatively impacted by placement of pipe whip restraints on certain systems. As a result, NRC and industry initiatives demonstrated that leak-before-break (LBB) criteria can be applied to RCS piping based on fracture mechanics technology and material toughness. Westinghouse performed the initial primary loop LBB analysis for HNP in 1984, which was documented in WCAP-10699 (Reference 2) and then approved by the NRC in 1985 (Reference 3).

In 1996 Westinghouse performed a LBB analysis for the replacement steam generator program using updated LBB criteria. The results were documented in WCAP-14549 (Reference 4).

5.5.2.2 Description of Analyses and Evaluations

Westinghouse has performed the LBB evaluation of HNP primary loop piping to justify the elimination of RCS primary loop pipe breaks with the Model Δ75 replacement steam generators at the uprated power of 2912.4 MWt. The evaluation was performed to demonstrate that the following objectives were achieved:

1. Demonstrate that margin exists between the critical crack size and a postulated crack that yields a detectable leak rate.

2. Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability of the HNP.

3. Demonstrate margin on applied load.

4. Demonstrate that fatigue crack growth is negligible.

The flaw stability criteria applied in the analysis considered both global and local stability for a postulated through-wall circumferential flaw. The global analysis was carried out using the plastic instability method, based on traditional plastic limit load concepts, but accounting for strain hardening, and taking into account the presence of a flaw. The local stability criteria were based on the J integral-tearing modulus (J-T) approach.

5.5.2.3 Acceptance Criteria

The LBB acceptance criteria are based on the Standard Review Plan 3.6.3. The recommended margins are as follows:

- Margin of 10 on the Leak Rate
- Margin of 2.0 on Flaw Size
- Margin of 1.0 on Loads (Using Faulted Load combination by Absolute Summation Method)

5.5.2.4 Results and Conclusions

Based on the evaluation performed, it is concluded that the LBB analysis results shown in WCAP-14549 (Reference 4) are acceptable for the HNP SGR/Uprating program. Therefore, the conclusions derived in References 2 and 4 remain valid, and the dynamic effects of the RCS primary loop pipe breaks need not be considered in the structural design basis of the HNP SGR/Uprating program.

Since the current evaluation has shown that the existing LBB analysis is acceptable for the SGR/Uprating program, the results continue to comply with the current HNP licensing basis/acceptance requirements. The results obtained with Model Δ75 replacement steam generators at the uprated power of 2912.4 MWt are also consistent with the results obtained with the Model Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

5.5.2.5 References

1. USNRC Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," February 1, 1984.

2. WCAP-10699, "Technical Justification for Eliminating Large Loop Pipe Rupture as the Structural Design Basis for the Shearon Harris Unit 1," September 1984.

3. Nuclear Regulatory Commission Docket No. 50-400; Letter from George W. Knighton, Chief Licensing Branch No. 3 Division of Licensing of the NRC to E. E. Utley, Executive Vice President Power Supply and Engineering and Construction of Carolina Power and Light Company, June 5, 1985.

4. WCAP-14549, "Technical Justification for Eliminating Large Loop Pipe Rupture as the Structural Design Basis for the Shearon Harris Unit 1 Nuclear Power Plant," December 1996.

5.5.3 Class 1 Auxiliary Lines

5.5.3.1 Introduction

The Harris Nuclear Plant (HNP) Class 1 Auxiliary Lines have been evaluated to determine the impact of the SGR/Uprating program design parameters on the qualification of the Class 1 Auxiliary Lines piping system.

5.5.3.2 Description of Analyses and Evaluations

The evaluation of the Class 1 Auxiliary Lines for the SGR/Uprating program include the following:

- The weight and center of gravity (CG) of the replacement steam generator
- The design transients for the SGR/Uprating program
- The NSSS parameters for the SGR/Uprating program
- Revised Class 1 Auxiliary Line pipe support stiffness values

All other input data and parameters other than those listed above remain the same as those used in Reference 1.

The six Class 1 Auxiliary Lines with pipe diameter sizes larger than 6 in. were reanalyzed for the SGR/Uprating program, since the existing seismic displacements at these large bore Class 1 Auxiliary Line connections to the RCL have changed and new seismic analysis for these lines was performed for the SGR. As a result, these six Auxiliary Lines were re-analyzed:

- Pressurizer Surge Line
- Accumulator Line Loop 1
- Accumulator Line Loop 2
- Accumulator Line Loop 3
- RHR Line Loop 1
- RHR Line Loop 3

These six lines were re-analyzed to assess the seismic impact of the SGR/Uprating on support loads and pipe stresses. The response spectra used is the same as that used for the existing design basis analysis. All other analyses, deadweight, thermal, LOCA, and fatigue were reconciled for the uprating conditions and the impact on the results were judged to be insignificant due to the uprating conditions.

The Class 1 Auxiliary Lines with pipe diameter sizes 6 in. and less were not re-analyzed for the SGR/Uprating program, since the existing seismic displacements at these small bore Class 1 Auxiliary Line connections to the RCL have not significantly changed and no new seismic analysis for these lines was performed for the SGR.

5.5.3.3 Acceptance Criteria

The acceptance criteria for the structural evaluation of the Class 1 Auxiliary Line piping system are provided in the design specification and procedure of References 2 and 3. As shown in Reference 2, the piping stress evaluation for the Class 1 Auxiliary Line piping stresses and the fatigue usage factors must be below the allowable limits as established in the ASME Code (References 4 and 5). The actual and allowable piping stresses and the actual and allowable cumulative usage fatigue factors are shown in Table 5.5.3-1

5.5.3.4 Results

The results of the evaluation for the six Class 1 Auxiliary Lines - Pressurizer Surge Line, Accumulator Line Loop 1, Accumulator Line Loop 2, Accumulator Line Loop 3, RHR Line Loop 1, and RHR Line Loop 3 are summarized in Table 5.5.3-1. The table lists the maximum Equation 9 Upset and maximum Equation 9 Faulted pipe stresses including the maximum cumulative fatigue usage factor in the Class 1 portion of the lines. The evaluation for the Class 1 Auxiliary piping determined the stresses and the fatigue usage factors to be below the allowable limits as established in the ASME Code (References 4 and 5).

The results of the remaining Class 1 Auxiliary Lines not listed in Table 5.5.3-1 remain unchanged from the original design basis for the SGR/Uprating program.

5.5.3.5 Conclusions

The HNP Class 1 Auxiliary Line piping has been evaluated for the SGR/Uprating program. The Class 1 Auxiliary piping, as stipulated in the design specification (Reference 2) is calculated to maintain its structural integrity and meet safety-related design requirements.

HNP operation with the Δ75 replacement steam generators at the uprated power of 2912.4 MWt bounds operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt. Operation at the uprated NSSS power is bounding since the associated NSSS parameters (e.g., maximum and minimum temperatures) and the design transients of concern for this evaluation result in calculated maximum stresses that remain acceptable with respect to the ASME criteria.

5.5.3.6 References

1. WCAP-9990 Volume IV, "Structural Analysis of Reactor Coolant Loop For Shearon Harris Nuclear Power Plant Unit 1 - Analysis Of The Class 1 Auxiliary Piping," November 1984.

2. Westinghouse Design Specification 955239, Revision 3, "Shearon Harris Nuclear Power Plant, Unit No. 1, Piping Design Specification, ANS Safety Class 1," and G-677458 Revision 4 plus Interim Revision 1 to G-677458 Revision 4, "General Piping Design Specification, ANS Safety Class 1."

3. CP&L Nuclear Generation Group Standard Procedure, Volume 99 Book/Part 99, EGR-NGGC-0308, "Pipe Stress Analysis," Revision 5.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB and NC, "Nuclear Power Plant Components," 1977 Edition and all addenda through Summer 1979, ASME, New York.

5. ASME Boiler and Pressure Vessel Code, Section III, Subsection NC, "Nuclear Power Plant Components," 1980 Edition, ASME, New York.

Table 5.5.3-1 Auxiliary Lines Class 1 Piping Stress Analysis Summary						
Auxiliary Line	**Maximum Equation 9 Upset Pipe Stress (KSI)**		**Maximum Equation 9 Faulted Pipe Stress (KSI)**		**Maximum Cumulative Usage Factor**	
	Actual Stress	**Allowable Stress**	**Actual Stress**	**Allowable Stress**	**Actual**	**Allowable**
RHR Loop 1	[ ]$^{(b,c)}$	28.03	[ ]$^{(b,c)}$	50.1	[ ]$^{(b,c)}$	1.0
RHR Loop 3	[ ]$^{(b,c)}$	28.03	[ ]$^{(b,c)}$	50.1	[ ]$^{(b,c)}$	1.0
Pressurizer Surge	[ ]$^{(b,c)}$	27.4	[ ]$^{(b,c)}$	48.2	[ ]$^{(b,c)}$	1.0
Accumulator Loop 1	[ ]$^{(b,c)}$	28.2	[ ]$^{(b,c)}$	50.1	[ ]$^{(b,c)}$	1.0
Accumulator Loop 2	[ ]$^{(b,c)}$	27.7	[ ]$^{(b,c)}$	50.1	[ ]$^{(b,c)}$	1.0
Accumulator Loop 3	[ ]$^{(b,c)}$	27.7	[ ]$^{(b,c)}$	50.1	[ ]$^{(b,c)}$	1.0

5.6 Reactor Coolant Pumps and Motors

The Harris Nuclear Plant (HNP) Model 93A Reactor Coolant Pumps (RCPs) were evaluated at SGR/Uprating conditions for structural adequacy (Section 5.6.1) and acceptability of the RCP motors (Section 5.6.2).

5.6.1 Reactor Coolant Pumps (Structural)

5.6.1.1 Introduction

This section addresses the structural adequacy of the HNP Model 93A RCPs at the SGR/Uprating conditions.

5.6.1.2 Description of Evaluations and Acceptance Criteria

The Westinghouse Model 93A RCPs were designed and analyzed to meet the HNP pump specifications (References 1 and 2) and the ASME Code of record (Reference 3). The SGR/Uprating seismic and LOCA loads remain bounded by the current operating conditions.

The evaluation of revised NSSS parameters indicates the RCP inlet temperature, which corresponds to the steam generator outlet temperature, is slightly lower, and thus bounded by the original design basis temperature. The normal operating pressure of 2250 psia remains the same as the original design basis for all of the cases. Therefore, the SGR/Uprating NSSS parameters are acceptable for the HNP RCPs.

The NSSS design transients were evaluated at SGR/Uprating conditions for any impact on the site specific and generic RCP analyses. The HNP RCP evaluation of the generic and site-specific pressure boundary summary reports showed that the fatigue analysis was performed using the Code [NB-3222.4(d)] waiver of fatigue requirements. The present HNP transients and the SGR/Uprating transients were compared.

The total number of Normal Condition transient cycles has decreased from the number required in Reference 1. A reactor coolant system (RCS) Cold Overpressurization transient was added to the original design basis upset transients. This transient is defined to occur 10 times. Each occurrence entails a thermal cycle along with an additional series of 600 pressure cycles. The RCS Cold Overpressurization transient is not a critical transient in the analysis of the RCP since it meets the ASME code fatigue waiver criteria. Therefore, the additional RCS Cold Overpressure transient cycles have no effect on the RCP fatigue evaluation.

The Normal Condition and Upset Condition transients are concerned with the temperature changes (ΔT) and the pressure changes (ΔP) related to the SGR/Uprating program. These parameters are of interest to the stress intensity range and fatigue considerations. In some cases,

the SGR/Uprating ΔP and/or the ΔT are equal to or less than the current E-Spec (Reference 1) values and are thus bounded and require no additional investigation.

For cases in which ΔP or ΔT increased, the effect on the various generic stress reports was evaluated. Since the generic reports use the ASME Code fatigue waiver, only a significant fluctuation of pressure or temperature is of interest in fatigue. None of the changes to the Normal or Upset transients for the SGR/Uprating cause a non-significant pressure or thermal transient to become a significant transient. The RCS Cold Overpressurization transient is also observed to be a non-significant transient for fatigue. Thus, the fatigue waiver evaluation for the current HNP licensing basis/acceptance requirements bounds the SGR/Uprating.

5.6.1.3 Results

The review of the effects of the SGR/Uprating NSSS parameters and NSSS design transients on the RCP site-specific and generic reports shows that the appropriate editions of record for the (Reference 3) are still met.

5.6.1.4 Conclusions

The revised NSSS parameters and NSSS design transients for the HNP SGR/Uprating are considered acceptable for the Model 93A RCPs from a structural standpoint. The pump pressure boundary parts continue to satisfy the RCP E-Specs (References 1 and 2) and the ASME Code of record (Reference 3). Therefore, the results of the structural evaluation of the HNP RCPs for the SGR/Uprating are consistent with and continue to comply with the current HNP licensing basis/acceptance requirements. The results obtained with the Δ75 replacement steam generators at the uprated NSSS power of 2912.4 MWt bound operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

5.6.1.5 References

1. Plant Design Specification 679138, Revision 6, "Carolina Power and Light Company Shearon Harris Nuclear Stations 1 & 2, Reactor Coolant Pump Model 93A, 60 Hertz," and Interim Revision No. 4, Westinghouse Nuclear Energy Systems, Pittsburgh, PA, D. R. Collier; Revised by C. T. Holmes and L. S. Jenkins, August 9, 1984.

2. General Equipment Specification G-677188, Revision 4, "Reactor Coolant Pump," Westinghouse Nuclear Energy Systems, Pittsburgh, PA, J. Green; Revised by A. A. Anderson, January 15, 1976.

3. "ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components," American Society of Mechanical Engineers, NY, 1971, with Addenda through the summer of 1972.

5.6.2 Reactor Coolant Pump Motors

5.6.2.1 Introduction

This section addresses the effect of the SGR/Uprating on the operation of RCP motors at the Harris Nuclear Plant (HNP) based on the NSSS parameters shown in Section 2.0 of this Licensing Report and a decrease in Best Estimate flow of 3400 gpm per loop.

5.6.2.2 Description of Evaluation and Acceptance Criteria

The seismic and LOCA loads remain bounded by the original requirements. Changes to the Component Cooling Water (CCW) system temperatures have an insignificant effect on motor operation. Due to insignificant effects from the SGR/Uprating transients, only the motor operation needs to be evaluated under the higher loads from the SGR/Uprating program. The effects of the SGR/Uprating on RCP motor components structural integrity, motor heat dissipation to the environment, and motor oil cooler temperature are insignificant.

The RCP motor performance is evaluated based on the maximum calculated hydraulic loads using the limiting NSSS parameters. The pump operating temperature of 536.3°F (steam generator outlet temperature) and the pump best estimate flow of 100,000 gpm are used to calculate the highest pump power requirements among the three RCPs at the HNP site. In addition, corresponding pump loads are calculated at cold condition (70°F).

All Westinghouse RCP motors are dual rated with a hot-loop rating (at which the NEMA Class B stator winding temperature rise limits are not to be exceeded) and a cold-loop rating (at which the NEMA Class F stator winding temperature rise limits are not to be exceeded). The motor does not have a service factor, but the cold-loop rating is 125 percent of the hot-loop rating.

The cold-loop rating is based on the water at 70°F. The hot-loop rating is based on the water temperature at the steam generator outlet during full-power operation. Both are considered to be steady-state cases.

5.6.2.2.1 Continuous Operation at Revised Hot-Loop Rating

The motor is required to drive the pump continuously under hot-loop conditions without exceeding a stator winding temperature rise of 75°C (corresponding to the NEMA Class B temperature rise limit in a 50°C ambient). The calculated hot-loop load is essentially at the motor nameplate rating of 7000 Hp. The motors have been shown by test to operate within the specification limits at the hot-loop nameplate rating. Therefore, continuous operation at the revised load is acceptable.

5.6.2.2.2 Continuous Operation at Revised Cold-Loop Rating

The motor is required to drive the pump for up to 3000 hours (continuous) under cold-loop conditions without exceeding a stator winding temperature rise of 100°C (corresponding to the NEMA Class F temperature rise limit in a 50°C ambient).

The worst-case cold-loop load under the revised operating conditions is 8907 Hp. This represents a 1.8-percent increase over the nameplate rating of the motor. The increase in stator winding temperature rise due to this new load is calculated to be 4.5°C above the NEMA guaranteed limit for Class F windings. Exceeding the Class F limits by 4.5°C during the 3000 hours of operation at cold conditions (70°F) can accelerate stator insulation thermal aging and reduce its life by approximately 1100 hours over the 40-year design life. Therefore, the motor cold operation time is reduced to 1900 hours over the 40-year RCP design life. With the increased loading, all available thermal margin has been utilized.

The analysis of the operation time at the cold condition is conservative.

5.6.2.2.3 Starting

The motor is required to start across the line with a minimum 80-percent starting voltage, against the reverse flow of the other pumps running at full speed, under cold-loop conditions. The limiting component for this type of starting duty is the rotor cage winding. A conservative all-heat-stored analysis is used to determine if the cage winding temperature exceeds the design limits (300°C on the bars and 50°C on the resistance rings).

Using a revised load torque curve, based on the SGR/Uprating conditions, the starting temperature rise for the rotor bars and resistance rings has been calculated. The results show bar temperature of 246.6°C and ring temperature of 32.95°C. These temperatures do not exceed design limits. Therefore, the motor can safely accelerate the load under worst case conditions.

5.6.2.2.4 Loads on Motor Bearings

Performance of the thrust bearings in an RCP motor can be adversely affected by excessive or inadequate loading. The axial down thrust for the revised parameters decreased from 55,000 lbs. to 53,604 lbs. for hot-loop operation and reduced from 75,000 lbs. to 72,251 lbs. for cold-loop operation. The thrust and radial bearing load changes that result from the SGR/Uprating program are well within the radial and thrust bearing capacities.

5.6.2.3 Conclusions

The worst case loads for the RCP motors were calculated for the SGR/Uprating conditions. Using the revised loads, all of the HNP RCP motors were evaluated in the four areas where parameter changes affect performance. These areas are as follows:

- Continuous operation at the revised hot-loop rating
- Continuous operation at the revised cold-loop rating
- Starting loads
- Bearing loads

Based on this evaluation, the RCP motors are considered acceptable for operation at the revised SGR/Uprating conditions.

The results for the SGR/Uprating are consistent with and continue to comply with the current HNP licensing basis/acceptance requirements. The results obtained with the Δ75 replacement steam generators at the uprated power of 2912.4 MWt bound operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

5.7 Steam Generators

The Harris Nuclear Plant (HNP) Model Δ75 replacement steam generators (RSGs) were analyzed at the uprated power conditions in the areas of structural acceptability (5.7.1), thermal-hydraulic (T/H) performance (5.7.2), and flow-induced vibration and wear (5.7.3). These three areas are discussed in detail below.

5.7.1 Steam Generator Structural Evaluation

5.7.1.1 Introduction

The uprating of HNP to 2912.4 MWt NSSS power incorporates steam generator tube plugging (SGTP) in the range of 0 percent to 10 percent maximum in any steam generator operating with the maximum feedwater temperature at either 440°F or 375°F, and a range of RCS temperature conditions in addition to the power increase. These conditions are considered in the RSG structural evaluations discussed below. The transients and load conditions for the RSG design specification are given in Reference 1.

The appropriate NSSS parameters were used for the steam generator structural integrity evaluation. In the evaluation, two sets of parameters were considered: high and low RCS temperatures. The enveloping condition for the primary side components is the one that results in a high primary to secondary side ΔP. The low temperature condition results in the lower steam pressure during normal operation. The lower pressure causes an increase in the corresponding primary to secondary ΔP, which results in higher stresses and slightly higher usage for the primary side components. For the secondary side components, the decrease in the maximum feedwater temperature and the resulting change in the temperature difference across some components can result in higher stresses and fatigue usage. Those components in direct contact with the feedwater and the subcooled fluid in the downcomer region are most affected by the change in feedwater temperature and are evaluated.

Those components that are within the steam space, or entirely within the confines of the steam generator, or whose fatigue usage factor is based on transients unaffected by the uprate, were not evaluated. These components are either not impacted by the change in the maximum feedwater temperature, since other transients not impacted by the uprate are controlling, or the thermal effects are not significant and so have negligible effect on the analysis results. The following components were not evaluated:

- Steam nozzle, elliptical head and upper shell
- Lower internals
- Sludge collector maintenance opening
- Upper internals

The NSSS design transients applicable for the uprating conditions were also used in this evaluation.

Feedwater line break (FWLB) effects have been considered both from the standpoint of the design pressures of the steam generator and the steam generator collapse pressure. Results show that the maximum external pressure exerted by the asymmetric loading of the FWLB is less than 6 percent of the pressure required to collapse the SG shell. Since the calculated collapse pressure is over-and-above the internal pressure of the SG, and the maximum applied sub-compartment load is less than 30 psi, the qualification of the SG is not impacted by this event. Therefore, FWLB asymmetric loading effects have been shown to be acceptable.

5.7.1.2 Description of Analyses and Evaluations

The structural evaluations were performed using the results of the previous structural analysis of the HNP Δ75 steam generators. Components not discussed below were not evaluated for the uprate, as discussed in Section 5.7.1.1, since the previous analysis results remain valid without further evaluation.

These evaluations were performed for the following SGR/Uprating primary and secondary side design conditions:

1. Uprated NSSS power of 2912.4 MWt
2. RCS vessel average temperature window of 572.0°F - 588.8°F
3. Model Δ75 replacement steam generators
4. Maximum SGTP level up to 10 percent
5. Full-power feedwater temperature of 440°F or 375°F
6. Maximum and minimum nominal steam pressure of 979 psia (0 percent SGTP) and 825 psia (10 percent SGTP), respectively.

The transients/NSSS parameters previously evaluated for the SGR were compared for differences that could affect the current evaluation. Conservative ratios, based on temperature differences, were used to modify the design analysis results, if required, to determine the stresses and usage factors at the SGR/Uprating conditions.

The structural evaluation of the critical components of the Model Δ75 steam generators was used as the basis to justify operation at the uprated conditions. The evaluations were performed to the requirements of the ASME B&PV code Section III, 1971 edition, Summer 1972 Addendum, Reference 2.

Material strength properties from the Summer 1972 Addendum to the 1971 ASME Code Section III were used except for those values not available in the 1971 Code. Where the material strength properties were not available in the 1971 ASME Code, Summer 1972 Addendum, material strength properties were taken from the 1986 Edition, Reference 3.

5.7.1.3 Acceptance Criteria

The critical steam generator components that were evaluated for structural adequacy are:

Primary side: Primary chamber, tubesheet, primary nozzles, primary manway, divider plate, and tube to tubesheet weld. The primary side of the replacement steam generators was evaluated as a whole through a review of the uprating transients that affect the primary side of the steam generator, i.e. RCS transients.

Secondary side: Upper shell, transition cone, lower shell, junction of tubesheet and stub barrel, main and auxiliary feedwater and spray nozzles, secondary manway opening and bolts, 6-in hand-hole, 2-in. and 4-in. inspection ports, and minor shell taps.

These components were evaluated for the effects of changes to the thermal transients due to the uprate. The acceptance criteria for each component is consistent with the criteria used in the design basis analysis referenced for each component.

5.7.1.4 Results

A summary of the results for each of these components follows. A summary of the maximum stress range/allowable stress range ratios and fatigue usage of the steam generator components is provided in Tables 5.7.1-1 and 5.7.1-2.

5.7.1.4.1 Primary Side Components

Primary side transients for uprating were compared to the applicable transients used in the design specification for the HNP RSGs. The comparison showed that the uprating transients for the RCS are the same as those used in the design specification (Reference 1).

Pressure conditions for the tubesheet are a function of both the primary and secondary side pressures. The primary side conditions are unchanged due to the uprating. The maximum primary-to-secondary side pressure difference would result from considering the maximum primary-side pressure and minimum secondary-side pressure. The minimum operational steam pressure specified for both the design basis analysis (Reference 1) and for SGR/Uprating (Section 2.1, Tables 2-5 and 2-6) is limited to 825 psia. Therefore, the design pressure difference of 1600 psia is not exceeded.

The current structural analysis of the SG tubes was reviewed. This analysis is based on an evaluation for the effects of:

- Deadweight,
- Tube – Tube sheet/flow distribution baffle/tube support plate interactions,
- Flow induced vibration and wear,
- Earthquake loadings,
- LOCA and pipe break transients, including LOCA rarefaction wave and shaking effects.

With the exception of flow-induced vibration and wear, which is addressed in Section 5.7.3 of this report, the other parameters are unaffected by the uprating. LOCA effects, which would show some impact due to uprate, were evaluated for the effects of a loop pipe break. Since leak-before-break has been applied to HNP, only the large auxiliary line breaks need to be considered. Therefore, the current analysis is conservative and will continue to envelop the LOCA effects in the uprate condition.

It can therefore be concluded that the effects of uprating on the primary side of the HNP RSGs has already been evaluated in the design basis analysis and remains valid for the SGR/Uprating conditions.

5.7.1.4.2 Secondary Side Components

For the analysis of the RSG secondary side components, the HNP design parameters were compared to those considered for the uprating. The major change identified is the change in the nominal feedwater temperature (either 375°F or 440°F). The other major change that affects the secondary side of the steam generator is the low-power feedwater introduction to the steam generator (either through the main or auxiliary nozzles). This change to the SG transients has been evaluated. The conclusion of the evaluation is that this change, while a more severe transient for some components, is enveloped by the current design basis analyses and has no impact on the currently reported stress levels and usage factors.

An evaluation of the effects of reducing the full-power feedwater temperature from 440°F to 375°F on the secondary-side thermal boundary conditions was performed. This evaluation used thermal-hydraulic boundary conditions such as pressure drops across internal components, heat transfer coefficients on internal surfaces and fluid pressures and temperatures for design and structural analysis of the steam generator. The evaluation was performed for the combined unit loading cases 0–15 percent and 15–100 percent power for each of the feedwater temperature case scenarios with the maximum feedwater temperature of either 375°F or 440°F.

The following subsections provide a summary of the stress analysis performed for each critical component in the secondary side of the steam generator. The calculated stresses are compared to

the allowable stress and stress range values calculated using S_m, S_y, and S_u values obtained from the ASME Code, Section III. For materials whose S_m, S_y, or S_u values are not available in the 1971 Edition of the Code, including Addenda through Summer of 1972, material strength properties are obtained from the ASME Code, Section III, 1986 Edition.

An evaluation for each component was performed to assess the impact on the HNP plant-specific component stress reports. In the subsections below, the maximum stress intensity ranges and fatigue usage factors associated with the Normal and Upset Loading Conditions are reported, if changed. All of the other loading condition stresses are unaffected by the uprate.

Main Feedwater, Auxiliary Feedwater and Spray Nozzles

An evaluation of the auxiliary feedwater, main feedwater, and spray nozzles was performed to address:

- Nozzle loads that exceed allowable values resulting from the reanalysis of the attached piping

- Changes to the design basis transients for the HNP SGR/Uprating

Results show that the design basis nozzle loads (Reference 1) remain applicable to the evaluation of the nozzles and produce stresses that envelop those produced by the mechanical loads. Thus, there is no impact on the current design basis analysis.

The evaluation of the changes to the design transients against the design analyses show that the major difference in the design transients due to the uprating is the result of considering a low maximum feedwater temperature of 375°F. This is in addition to the high maximum feedwater temperature of 440°F that has already been evaluated. All other differences in the transients are not significant and will not impact the current design basis analysis.

The evaluation of the steam generator feedwater and spray nozzles for the lower maximum feedwater temperature of 375°F show that the current design basis analysis contains sufficient conservatism to account for any increase in stress that may result from the lower maximum feedwater temperature. Also, the transients that contribute most to the fatigue usage factor for the components are unaffected by the uprating. Therefore the effect of changes in the uprating transients from those considered in the design basis analysis is further minimized.

The conclusion is that the design basis analysis for the main and auxiliary feedwater nozzles and the spray nozzles remains acceptable for SGR/Uprating conditions.

Secondary Manway Opening

The HNP Δ75 steam generator contains two secondary-side manways, which are located in the upper shell, 180 degrees apart. The manway is a forged component welded to the shell opening. A manway cover and insert are fastened to the pad with studs and nuts, and a spiral wound graphite filled gasket is used to seal the joint.

An evaluation for the effects of uprating was performed. The primary stresses for the pad and cover, and the bolt sections, are within the applicable limits of the ASME Code. The engagement lengths and the bolt areas are adequate. The secondary manway sections were analyzed for maximum stress intensity range and fatigue. The maximum stress range/allowable stress range ratios and fatigue usage factors for the baseline and uprated cases are indicated in Tables 5.7.1-1 and 5.7.1-2. All the calculated fatigue usage factors are below unity and are acceptable, with the exception of the bolts, which demonstrate a cumulative usage factor of greater than 1.0 over a 40-year life. Based on conservative evaluations, the bolts must be replaced prior to 27 years of service to meet the fatigue usage of 1.0. The bolts must be replaced as part of normal maintenance activities, whenever it is convenient, prior to the 27th year of the operating period. If some of the bolts are replaced for reasons other than fatigue, and records are available which indicate when and which bolts have been replaced, those bolts would not need to be replaced again as long as they did not exceed the 27-year operation limit.

Hand-Hole, 6.00 Inch

The Δ75 steam generator contains six hand-holes located in the secondary shell directly above the tubesheet. These hand-holes permit 6-inch access to the tube bundle.

An evaluation for the effects of uprating was performed. The primary stresses for the shell/pad and cover, and the bolt sections are within the applicable ASME Code limits. The engagement lengths and bolt areas are adequate. All the calculated fatigue usage factors are below unity. The hand-hole sections were analyzed for maximum stress intensity range and fatigue. The maximum stress range/allowable stress range ratios and fatigue usage appear in Tables 5.7.1-1 and 5.7.1-2, respectively. While the maximum primary plus secondary stress intensity ranges are unaffected by the uprating, the fatigue usage factors are affected. The maximum usage factor is less than the ASME Code allowable of 1.0 and is therefore acceptable.

Inspection Port, 4.00 Inch Bolted Closure

The 4-in. bolted closure inspection port for the Δ75 steam generator is located at the lower shell to transition cone junction, which allows visual access to the tubes at the top tube support plate. The inspection port is exposed to down-comer fluid temperature changes and secondary-side pressure fluctuations.

An evaluation for the effects of uprating was performed. The primary stresses for the pad and cover, and the bolt sections are within the applicable limits of the ASME Code. The engagement lengths and the bolt areas are adequate. The inspection port sections were analyzed for maximum stress intensity range and fatigue. The maximum stress range/allowable stress range ratios and fatigue usage factors for the baseline and uprated cases are indicated in Tables 5.7.1-1 and 5.7.1-2. All the calculated fatigue usage factors are below unity and are acceptable with the exception of the bolts. Based on conservative evaluations, the bolts will need to be replaced prior to 30 years of operation. If some of the bolts are replaced for reasons other than fatigue, and records are available which indicate when and which bolts have been replaced, those bolts would not need to be replaced again as long as they did not exceed the 30-year operation limit.

Inspection Port, 2.00 Inch Unreinforced

The Δ75 steam generator contains sixteen 2-in. inspection ports. These are located in the lower shell approximately 3 inches above the tube support plates B-J. The inspection ports directly above the tube support plate B are located in the reinforced area of the stub barrel, while all the other inspection ports are located in the lower shell and are not reinforced. The following analyses apply to the unreinforced 2-in. inspection ports of the HNP RSGs.

An evaluation for the effects of uprating was performed. The primary stresses for the shell and cover, and the bolt sections are within the applicable limits of the ASME Code. The engagement lengths and the bolt areas are adequate. The inspection port sections were analyzed for maximum stress intensity range and fatigue. The maximum stress range/allowable stress range ratios and fatigue usage factors for the baseline and uprated cases are indicated in Tables 5.7.1-1 and 5.7.1-2. All the calculated fatigue usage factors are below unity and are therefore acceptable.

Minor Shell Taps

The HNP Δ75 steam generator shell has several drilled penetrations that provide ports for instrumentation and small-bore piping connections. A counter bore into the shell provides a weld-prep for the bosses that are welded to the shell. The bosses provide an internally machined pipe connection for the water-level instrumentation system, the wet lay-up system, the blowdown system, and the secondary-side drain system.

An evaluation for the effects of uprating was performed. The primary stresses for the critical sections are within the applicable limits of the ASME Code. The minor shell tap sections were analyzed for maximum stress intensity range and fatigue. The maximum stress range/allowable stress range ratios and fatigue usage factors for the baseline and uprated cases are indicated in Tables 5.7.1-1 and 5.7.1-2. The fatigue usage factor at the most critical section was acceptable.

Tubesheet/Stub Barrel Junction

The critical region is the weld between the tubesheet and the stub barrel. The effect of a change in feedwater temperature from 440°F to 375°F was evaluated

The stresses were scaled based on a ratio of the temperature differences in the lower downcomer region. The resulting stress intensity ranges were compared against the ASME Code allowable of $3S_m$. The stress intensity ranges remain acceptable. The scaling factor was conservatively applied to all load combination thermal alternating stresses for the critical region. The maximum usage factor is less than 1.0 and is acceptable. The maximum stress range/allowable stress range ratios and fatigue usage factors are indicated in Tables 5.7.1-1 and 5.7.1-2.

Upper Shell/Transition Cone/Lower Shell

The evaluation of the upper shell, transition cone and lower shell regions was performed to address temperature differences resulting from a full-power feedwater temperature reduction of 440°F to 375°F.

The feedwater temperature affects only the following transients:

- Loading/unloading from 0–100 percent loading
- 10-percent step load increase/decrease
- Large step load w/steam dump
- Loop out-of-service loop startup/shutdown
- Reactor trip cases (A, B and C)
- A large feedwater line break

A scaling factor due to the feedwater temperature change was applied conservatively to all thermal alternating stresses for all of the thermal transients in the fatigue evaluation.

The resulting stress intensity ranges were compared against the Code allowable of $3S_m$. For those stresses that exceeded the Code limit, an analysis was performed per paragraph NB-3228.3 of the ASME Code, and the conditions were satisfied. The results of the evaluation are presented in Tables 5.7.1-1 and 5.7.1-2. The maximum fatigue usage factor is less than the allowable of 1.0 and is acceptable.

5.7.1.5 Conclusions

A summary of the maximum stress range/allowable stress range ratios and fatigue-usage factors for the various steam generator components evaluated/analyzed is provided in Tables 5.7.1-1 and

5.7.1-2. Results of the analyses performed on the HNP Model Δ75 steam generators show that ASME Code Section III is satisfied at the uprated power conditions with up to 10-percent steam generator tube plugging, with the exception of the secondary manway bolts and 4-in. inspection port bolts. For these bolts it was found that the maximum fatigue usage exceeds the limit of 1.0. Therefore, these bolts will be replaced at a convenient time as part of normal maintenance activities, prior to the bolts reaching the stated fatigue life of 27 and 30 years of service for the manway and 4-in. inspection port bolts, respectively.

The results of the evaluation for the Δ75 RSGs at the uprated NSSS power of 2912.4 MWt bound operation with the Δ75 RSGs at the current NSSS power of 2787.4 MWt.

5.7.1.6 References

1. Design Specification 412A86, Revision 4, "Carolina Power and Light Company HNP Delta 75 Steam Generator," October 23, 1998.

2. 1971 ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, including Summer 1972 Addenda.

3. 1986 ASME Boiler and Pressure Vessel Code, Section II, Materials.

Table 5.7.1-1 Summary of Maximum Stress Range/Allowable Stress Range Ratios in Steam Generator Components		
Components	**HNP Baseline**	**Uprating Lower FW Temperature**
Primary Side Components (Max. of all Components)	(1)	(1)
Main Feedwater	(1)	(1)
Auxiliary Feedwater	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Spray Nozzles	(1)	(1)
Secondary Manway Opening	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Hand-Hole	(1)	(1)
4-in. Inspection Port	(1)	(1)
2-in. Un-reinforced Inspection Port	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Minor Shell Taps	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Tubesheet/Stub Barrel Junction	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Upper Shell/Transition Cone/Lower Shell	[ ]$^{(b,c)}$	(1)

Note:

(1) Exceeded the $3S_m$ allowable criteria. Per paragraph NB-3228.3 of the ASME Code, a simplified elastic-plastic analysis was performed, and all Code requirements were met.

Table 5.7.1-2 Summary of Fatigue Usage of Steam Generator Components		
	Fatigue Usage	
Components	HNP Baseline	Uprating Lower FW Temperature
Primary Side Components (Max. of all Components)	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Main Feedwater	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Auxiliary Feedwater	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Spray Nozzles	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Secondary Manway Opening	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Secondary Manway Bolts	[ ]$^{(b,c)}$	[ ]$^{(b,c)(1)}$
Hand-Hole	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Hand-Hole Bolts	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
4-in. Inspection Port	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
4-in. Inspection Port Bolts	[ ]$^{(b,c)}$	[ ]$^{(b,c)(2)}$
2-in. Un-Reinforced Inspection Port Opening	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
2-in. Un-Reinforced Inspection Port Bolts	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Minor Shell Taps	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Tubesheet/Stub Barrel Junction	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Upper Shell/Transition Cone/Lower Shell	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$

Notes:

(1) Bolts must be replaced prior to 27 years of service.

(2) Bolts must be replaced prior to 30 years of service.

5.7.2 Steam Generator Thermal-Hydraulic Evaluation

5.7.2.1 Introduction

The Model Δ75 replacement steam generator installed for HNP has been designed and evaluated at the uprated power conditions, 2912.4 MWt (970.8 MWt/loop). The thermal-hydraulic design for these steam generators evaluated multiple operating points at the uprated power level as well as the original HNP power level, 2787 MWt (929 MWt/loop). The SGR/Uprating program thermal-hydraulic evaluation considers only the effect of reducing feedwater temperature from 440°F to 375°F. Evaluations at 440°F were repeated for the SGR/Uprating program, as the current parameter assumptions in Section 2 of this NSSS Licensing Report are slightly different.

Applicable design parameters for operation at the current and reduced feedwater temperatures were used for the thermal-hydraulic evaluation. The operating steam generator water level was set at the normal level, 520 inches above the tubesheet surface. The fouling factor was taken at the design value for all conditions.

5.7.2.2 Description of Analyses and Evaluations

The eight parameter cases used for the steam generator thermal-hydraulic evaluations for the SGR/Uprating Project are defined in Section 2 of this NSSS Licensing Report.

The thermal-hydraulic evaluation of the steam generators focused on secondary side operating characteristics. Comparisons with the conditions at the current design feedwater temperature are used to demonstrate the acceptability of the characteristics at the reduced feedwater temperature. Special attention is given to hydrodynamic stability margin that decreases as a result of the operating condition changes.

5.7.2.3 Acceptance Criteria

Several secondary-side operating characteristics were used to assess the acceptability of steam generator operation at reduced feedwater temperature. These parameters include steam flow and pressure; circulation ratio; damping factor, which is a measure of hydrodynamic stability; secondary mass; heat flux; and secondary side pressure drop. Moisture carryover is also affected and is addressed.

5.7.2.4 Results

5.7.2.4.1 Thermal-Hydraulic Operating Characteristics

Secondary-side steam generator performance characteristics can be affected by changes in steam flow and steam pressure. The current HNP evaluations are performed for uprated power conditions. The primary temperature and SGTP level determine steam pressure. Steam flow at

constant power is affected by feedwater temperature changes. This section assesses the magnitude and importance of changes in the secondary-side thermal-hydraulic performance characteristics that will result from a reduction in feedwater temperature.

Most of the operating characteristics displayed expected small or benign changes as a result of a reduction in the feedwater temperature. Steam flows decrease for the reduced feedwater temperatures as a result of the increased enthalpy difference at constant power. The steam pressures are not affected by feedwater temperature. Circulation ratio increases proportionally to the decrease in steam flow as is typical for re-circulating steam generators. Bundle liquid flow is related to clearing and reducing the concentration of contaminants in the bundle. The increased bundle liquid flows, at reduced feedwater temperature, will improve the effectiveness of this useful function.

At a given power, average heat fluxes are proportional to the heat transfer area in service and are not affected by the feedwater temperature. Peak heat fluxes are similarly unaffected by feedwater temperature. Total secondary-side pressure drop is smaller as a result of the reduced steam flows. Steam generator mass increases with reduced feedwater temperature. The reduced steam flow simulates reduced power with its attendant reduction in tube bundle void.

5.7.2.4.2 Moisture Carryover

Moisture separator performance benefits from the reduced steam flows. Steam flow usually dominates separator loading and therefore performance. This is especially true in the present case since the other factors that affect moisture carryover, steam pressure and water level, remain constant. The evaluation results indicated a reduced separator loading with improved performance and lower moisture carryover at the low feedwater temperature.

5.7.2.4.3 Hydrodynamic Stability

The hydrodynamic stability of a steam generator is characterized by the damping factor. A negative value of this parameter indicates a stable unit. That is, small perturbations of steam pressure or circulation ratio will die out rather than grow in amplitude. A number of factors can have a destabilizing effect on steam generators. Two of these factors are increased downcomer sub-cooling and increased pressure drop in the two-phase region. For the HNP the reduced stability is caused by increased downcomer sub-cooling, resulting from colder feedwater added to the downcomer.

For the reduced feedwater temperature cases, the damping factor increases (becomes less negative) in relation to the value for the 440°F feedwater temperature. The damping factors at reduced feedwater temperature, however, remain substantially negative. It can therefore be inferred that the HNP units will continue to be hydrodynamically stable.

5.7.2.5 Conclusions

All projected thermal-hydraulic operating characteristics are acceptable for both the 375°F and 440°F feedwater temperature and SGR/Uprate operating conditions. Though the evaluations were performed at the uprated power of 2912.4 MWt, this conclusion is also applicable to the current NSSS power of 2787.4 MWt. The results are compatible with the current HNP licensing basis/acceptance requirements.

The effect of feedwater temperature reduction was shown to have a small or benign effect on thermal-hydraulic operating characteristics for a wide range of operating conditions at the uprated power, 2912.4 MWt. The changes observed for the feedwater temperature reduction evaluated are consistent with experience and are expected to be similar at other operating conditions that are not far from the uprated conditions. On this basis, the current rating, about 96 percent of the uprated power level, will show similarly small or benign effects due to feedwater temperature changes.

5.7.3 Flow Induced Vibration and Wear

5.7.3.1 Introduction

The tube bundle support assembly is primarily a Safety Class 3 structure (support plates and anti-vibration bars [AVBs]) with some parts (wrapper barrels, AVB end caps, retaining rings, and retainer bars) classified as non-nuclear safety (NNS) according to ANSI/ANS N18.2a (Reference 1). The tubes comprise part of the Class 1 primary-side pressure boundary, which must satisfy ASME Code limits consistent with the original design basis, Reference 2. The original vibration/wear report assessed vibration and wear analyses and corrosion/erosion conditions, considering interaction with the flow distribution baffle (FDB), nine tube support plates (TSPs), and four sets of AVBs. The resulting bending stresses and the maximum potential for localized wear due to tube vibration were inputs to the original structural evaluation of the tube. The structural evaluation of the tubes evaluates the effects of pressure and temperature resulting from the transients. The tube structural evaluation also considers the:

- Deadweight
- Tube – Tube sheet/flow distribution baffle/tube support plate interactions
- Earthquake loadings
- LOCA and pipe break transients, including LOCA rarefaction wave and shaking effects

These effects are discussed in Section 5.7.1 of this report.

The only significant change identified as a result of the SGR/Uprating is the reduced feedwater temperature which can be maintained at an upper temperature at full power of 375°F rather than the 440°F, as specified in the Model Δ75 design specification. An evaluation was performed to determine if tube vibration, which can affect wear rates and tube fatigue, is affected by the 375°F feedwater temperature option specified for the SGR/Uprating.

5.7.3.2 Description of Analyses and Evaluations

An evaluation of the thermal-hydraulic boundary conditions was performed for the heatup transient from 0-percent to 100-percent power for each scenario of maximum feedwater temperature, 375°F and 440°F. The results of the evaluation were used in evaluating the impact on flow-induced vibration on the original analysis.

5.7.3.3 Results

The original analysis case differs from the SGR/Uprating case only in terms of the feedwater temperature reduction.

The current evaluation is based on the following:

- The original analysis evaluation (440°F) concluded that tube wear would be relatively benign.

- The wear and flow-induced vibration conditions for the lower feedwater temperature conditions (375°F) show a reduction in the wear related parameters.

5.7.3.4 Conclusions

It is concluded that flow induced vibration and wear for the 375°F feedwater case is acceptable since the previous analysis for the 440°F feedwater case envelops the 375°F feedwater case. It can further be concluded that the previous structural analysis of the tubes remains bounding relative to the SGR/Uprating since the input parameters related to flow-induced vibration are unchanged.

This evaluation at the uprated power of 2912.4 MWt bounds operation with the replacement steam generators at the current NSSS power of 2787.4 MWt.

5.7.3.5 References

1. "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI/ANS N.18.2a, 1975

2. ASME Boiler and Pressure Vessel Code Section III, "Rules for Construction of Nuclear Power Plant Components," 1971 Edition plus Addenda through Summer 1972 and Case N-20-3, The American Society of Mechanical Engineers, New York, New York

5.8 Pressurizer

5.8.1 Introduction

The functions of the pressurizer are to absorb any expansion or contraction of the primary reactor coolant due to changes in temperature and/or pressure and, in conjunction with the pressure control system components, to keep the reactor coolant system (RCS) at the desired pressure. The first function is accomplished by keeping the pressurizer approximately half-full of water and half-full of steam at normal conditions, connecting the pressurizer to the RCS at the hot leg of one of the reactor coolant loops and allowing inflow to or outflow from the pressurizer as required. The second function is accomplished by keeping the temperature in the pressurizer at the water saturation temperature (T_{sat}) corresponding to the desired pressure. The temperature of the water and steam in the pressurizer can be raised by operating electric heaters at the bottom of the pressurizer and can be lowered by introducing relatively cool spray water into the steam space at the top of the pressurizer.

The components in the lower end of the pressurizer (such as the surge nozzle, lower head/heater well and support skirt) are affected by pressure and surges through the surge nozzle. The components in the upper end of the pressurizer (such as the spray nozzle, safety and relief nozzle, upper head/upper shell, manway and instrument nozzle) are affected by pressure, spray flow through the spray nozzle, and steam temperature differences.

The limiting operating conditions of the pressurizer occur when the RCS pressure is high and the RCS hot-leg (T_{hot}) and cold-leg (T_{cold}) temperatures are low. This maximizes the ΔT that is experienced by the pressurizer. Due to flow out of and into the pressurizer during various transients, the surge nozzle alternately sees water at the pressurizer temperature (T_{sat}) and water from the RCS hot leg at T_{hot}. If the RCS pressure is high (which means, correspondingly, that T_{sat} is high) and T_{hot} is low, then the surge nozzle will see maximum thermal gradients; and, thus experience the maximum thermal stress. Likewise, the spray nozzle and upper shell temperatures alternate between steam at T_{sat} and spray water, which, for many transients, is at T_{cold}. Thus, if RCS pressure is high (T_{sat} is high) and T_{cold} is low, then the spray nozzle and upper shell will also experience the maximum thermal gradients and thermal stresses.

The original evaluation of the Harris Nuclear Plant (HNP) pressurizer was performed in Reference 1 considering the vessel operating temperatures, design transients and design loads in References 2 and 3. The design transients in References 2 and 3 are the design transients from Systems Standard Design Criteria (SSDC) 1.3, Rev. 2 (Reference 4) except for several special HNP transients (i.e., Plant Loading and Unloading at 10-Percent per Minute and 20-Percent Step Load Increase and Decrease). Therefore, References 1 through 4 document the original HNP pressurizer design basis. This evaluation was updated in Reference 5 to include the Cold Overpressure Mitigation System (COMS) transient added to the plant specific design specification in Reference 6.

The pressurizer stress report was updated again in 1993 (Reference 7) in order to incorporate revised operating temperatures and NSSS design transients associated with the T-Hot Reduction/Replacement Steam Generator (RSG) Program into the pressurizer design. Reference 7 contained the evaluations necessary to justify pressurizer operation with T_{cold} temperatures within a range from 536.6°F to 555.5°F, and T_{hot} temperatures within a range from 605.9°F to 623.2°F. The evaluations in Reference 7 also considered the surge line stratification loads and the revised NSSS design transients, which were applicable to the respective high T_{avg} and low T_{avg} programs, that were developed to cover the entire operating temperature window. The evaluations documented in Reference 7 were performed using the most conservative design inputs so that pressurizer operation in accordance with the HNP T-Hot Reduction/RSG Program, the original design basis, or a combination thereof is acceptable for the remainder of the operating license. The HNP pressurizer was subsequently reviewed for the SGR/20-Percent SGTP and Operating Temperature Range Program (Reference 8) in 1996. That report stated that the T-Hot Reduction/Replacement Steam Generator (RSG) Program analysis (Reference 7) remains bounding for the RSG case.

The HNP SGR/Uprating program requires additional pressurizer structural analyses and evaluations for the purpose of incorporating hot and cold-leg temperatures, NSSS design transients and design loads that are not bounded by the current HNP pressurizer stress report and addenda.

5.8.2 Description of Analyses and Evaluations

The evaluation of the pressurizer for the SGR/Uprating used the following as input:

- NSSS Parameters (Section 2.0)
- NSSS Component Design Transients (Section 3.0)
- Pressurizer Spray (up to 1200 gpm)
- Design Loads
- Pressurizer Nozzle and Support Skirt Loads

The input parameters associated with the HNP SGR/Uprating program were reviewed and compared to the design inputs considered in the current pressurizer stress report. In cases where revised input parameters are not obviously bounded, pressurizer structural analyses and evaluations were performed. Any impacts to the existing design basis analysis were evaluated. Using the existing analyses as the basis of the evaluation, scaling factors were utilized to assess the impact of the changes in the parameters such as the system transients, temperatures, and pressures. New stresses and revised cumulative usage factors were calculated, as applicable, and compared to previous licensed results. The evaluation results were then compared with the ASME Code (Reference 9) to confirm that the allowable limits are maintained.

A review of the design transients shows that some of the transients have been revised, although not all parameters affecting the pressurizer have been revised for each transient. The number of occurrences for the SGR/Uprating transients are identical to those considered in the pressurizer specification (Reference 6).

The review of the pressure fluctuations showed that the differences are very small and will not have any significant effect on the stress analysis and fatigue evaluation of the pressurizer components that were originally analyzed in Reference 5.

The ΔTs between the pressurizer and the incoming T_{cold} were reviewed. The ΔTs listed for the stress report volumes (Reference 10) were the values used in the original analyses. For the spray nozzle, the ΔTs for all transients are less than or equal to the ΔTs used in the original analyses. The spray flow rates, however, are higher in the SGR/Uprating program for the heatup and cooldown transients (which have the same ΔTs as before), as well as for some of the other transients that have lower ΔTs. Of the equipment design transients that have lower ΔTs, the Loss-of-Load and Inadvertent Startup of an Inactive Loop would result in higher stresses in the spray nozzle than were calculated originally.

For the upper shell and trunnion buildup, the only transients that affect these locations are plant heatup and cooldown. Flow rate and ΔT comparisons for transients other than heatup and cooldown, therefore, have no bearing on the fatigue calculations for these components.

The ΔTs between the pressurizer and the incoming T_{hot} were reviewed. The ΔTs listed for the stress report volumes (Reference 10) were the values used in the original analyses. The lower head/heater well and immersion heaters' ΔTs used in the stress report are higher than those for the revised transients. Therefore, the stress analysis of these components does not need to be updated. For the surge nozzle, the lower head/support skirt, and instrument nozzle, some of the revised transients have higher ΔTs than those used in the original stress report volumes. The analyses for these components were updated.

The review of the Emergency and Faulted transient temperatures showed that there were no changes that would affect the pressurizer.

The design loads considered in the pressurizer structural analysis documented in Reference 7 remain conservative for the HNP SGR/Uprating program. Reconciliation of the pressurizer nozzle and support skirt loads with the generic design loads was done as part of the loop analysis effort. The nozzle loads and support skirt loads are design inputs for the pressurizer structural analysis only if additional load conformance or stress calculations become necessary to show compliance. The pressurizer nozzle and skirt loads were shown to remain limiting.

5.8.3 Acceptance Criteria

The initial set of acceptance criteria for evaluating design inputs affecting the pressurizer stress report are as follows:

1. Hot and cold-leg temperatures shall remain within the ranges of the operating temperatures that have previously been considered and justified in the pressurizer stress report. (The effect of the higher spray flow rate at HNP is incorporated in the comparisons for T_{cold}.)

2. NSSS design transients shall be less than or equal to the design transients previously considered in the pressurizer stress report with regard to both severity and numbers of occurrences. Additionally, no new NSSS design transients that have not previously been considered shall be identified. (The pressurizer temperature and pressure variations for each transient shall be considered in this comparison review to determine the relative severity of the revised design transients compared to the existing design transients.

3. Design loads shall be less than or equal in magnitude to the loads that were previously considered in the pressurizer stress report with no changes to the load application points and numbers of occurrences.

The acceptance criteria for the pressurizer structural analyses and evaluations are in accordance with the applicable requirements of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code with addenda through the Summer 1972 Addenda (Reference 9) to which the HNP pressurizer was originally designed.

5.8.4 Results

The analysis of the pressurizer for the HNP SGR/Uprating program transients will have a minimal effect on the pressurizer components. Table 5.8-1 compares the fatigue usages calculated for the SGR/Uprating with those of the T-Hot Reduction/RSG Program. The T-Hot Reduction/RSG Program forms the current analysis basis of the pressurizer for HNP (Reference 7). The fatigue usages for the upper shell and trunnion bolt hole decreased significantly due to the removal of excessive conservatism in the original evaluation. For components with no change in fatigue usage factor in Table 5.8-1, the existing analyses envelop the revised transients. From this analysis, it is concluded that the HNP pressurizer components satisfy the requirements of the ASME Code for the revised transients of the SGR/Uprating.

5.8.5 Conclusions

The results of the pressurizer analysis show that the HNP pressurizer components meet the stress/fatigue analysis requirements of the ASME Code, Section III (Reference 9) at the SGR/Uprating conditions.

The results are consistent with and continue to comply with the current HNP licensing basis/acceptance requirements. The results obtained with the Δ75 replacement steam generators at the uprated NSSS power of 2912.4 MWt bound operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

5.8.6 References

1. "Model D Series 84 Pressurizer Stress Report for Carolina Power and Light Shearon Harris Unit 1," WNET-130 (CQL), Rev. 1, Westinghouse Electric Corporation, Tampa Division, Tampa, Florida, January 1984.

2. Equipment Specification 676440, Revision 5, "General Pressurizer Vessel Assembly Series 84," Westinghouse Electric Corporation, Nuclear Energy Systems, Pittsburgh, PA, October 1973.

3. Equipment Specification 952407, Revision 5, "Addendum to Equipment Specification 676440, Revision 5, Carolina Power and Light, Shearon Harris Nuclear Plant Unit 1 Pressurizer," Westinghouse Electric Corporation, Nuclear Energy Systems, Pittsburgh, PA, November 1983.

4. Westinghouse System Standard Design Criteria (SSDC) 1.3, Rev. 2, April 15, 1974.

5. "Model D Series 84 Pressurizer Stress Report for Carolina Power and Light Shearon Harris Unit 1," WNET-130 (CQL), Rev. 2, Westinghouse Electric Corporation, Tampa Division, Tampa, Florida, August 1989.

6. Equipment Specification 952407, Revision 6, "Addendum to Equipment Specification 676440, Revision 5, Carolina Power and Light, Shearon Harris Nuclear Plant Unit 1 Pressurizer," Westinghouse Electric Corporation, Nuclear Energy Systems, Pittsburgh, PA, January 1989.

7. "Shearon Harris (CQL) Pressurizer Evaluation for Uprating/T-Hot Reduction/Steam Generator Replacement Program," EA-93-260, Westinghouse Electric Corporation, Pensacola Plant, Pensacola, FL, December 7, 1993.

8. WCAP-14778, "Carolina Power and Light, Shearon Harris Nuclear Power Plant Steam Generator Replacement with 20% Steam Generator Tube Plugging and Operating Temperature Range Engineering Report," Westinghouse Electric Corporation, Nuclear Energy Systems, Pittsburgh, PA, December, 1996.

9. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," 1971 Edition through the Summer 1972 Addenda.

10. Model D Series 84 Pressurizer Generic Stress Report, WNET-130 Volumes 2–16, Westinghouse Electric Corporation, Tampa Division, Tampa, Florida, 1977 (Westinghouse Proprietary Class 1 Document).

Table 5.8-1 HNP Fatigue Usage Comparisons		
Component	**SGR/Uprating Fatigue Usage**	**T-hot Red/ RSG Fatigue Usage**
Surge Nozzle	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Spray Nozzle – Safe End	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Spray Nozzle – Nozzle Body/Shell Intersection	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Safety and Relief Nozzle	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Lower Head	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Heater Well	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Upper Shell	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Support Skirt/Flange	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Support Skirt/Lower Shell	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Seismic Lug at Lug/Shell Junction	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Manway Cover/Bolts	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Trunnion Bolt Hole	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Instrument Nozzle	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Immersion Heater/Sheath	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$
Valve Support Bracket	[ ]$^{(b,c)}$	[ ]$^{(b,c)}$

5.9 NSSS Auxiliary Equipment

5.9.1 Introduction

This report evaluates the Harris Nuclear Plant (HNP) auxiliary tanks, heat exchangers, pumps and valves, on a system basis, for impact by the thermal transients and maximum operating temperatures, pressures, and flow rates resulting from the SGR/Uprating conditions. The evaluation consists of a structural and flow-capacity review of the component pressure boundaries.

The NSSS Auxiliary Systems associated with this evaluation include:

- Gaseous Waste Processing System (GWPS)
- Chemical and Volume Control System (CVCS)
- Boron Thermal Regeneration System (BTRS)
- Residual Heat Removal System (RHRS)
- Safety Injection System (SIS)
- Boron Recycle System (BRS)
- Component Cooling Water System (CCWS)
- Liquid Waste Processing System (LWPS)

5.9.2 Description of Analyses and Evaluations

The PCWG parameters provided in Section 2 document the impact of the SGR/Uprating on the NSSS system operating temperatures and pressures. This information was applied where applicable for evaluation of the auxiliary equipment maximum operating temperatures and pressures. Any impacts on the NSSS and auxiliary equipment design transients due to the SGR/Uprating conditions were considered.

The original design parameters (design temperature, pressure, thermal transients and flow rates) for the auxiliary tanks, heat exchanger, pumps and valves, were compared to those used in the SGR/Uprating to determine if the original design parameters envelop those for the SGR/Uprating.

5.9.2.1 Auxiliary System Tanks

The NSSS auxiliary tanks that were evaluated are listed in Table 5.9-1. From an evaluation of the revised SGR/Uprating conditions, the operating temperatures and pressures for these vessels remain within the design basis for these tanks, and the equipment design transients for the boron

injection tank and the safety injection accumulators remain bounded by the original design transients. No design transients were identified for Auxiliary System Tanks other than the boron injection tank and the safety injection accumulators. As a result, none of the auxiliary tanks are impacted and are therefore acceptable for operation at the SGR/Uprating conditions.

Although the CCW System is subject to higher operating temperatures under the uprated conditions, these operating conditions are bounded by the original design conditions.

The pressurizer relief tank sparger was not included in this evaluation because it is not impacted by either the SGR or Uprating.

5.9.2.2 Auxiliary System Heat Exchangers

The NSSS auxiliary heat exchangers that were evaluated are listed in Table 5.9-2. Several heat exchangers have two systems listed. This is because the tube-side and shell-side circulate water from two different systems.

The CCW system is subject to higher operating temperature conditions as a result of the SGR/Uprating. All of these temperatures remain bounded by the original design conditions.

New flow rates were defined as a result of the SGR/Uprating. Westinghouse has reviewed the increased flow rates against the original design calculations. The design specification includes criteria defined to prevent flow-induced vibration at twice the design flow. The increased flow rates are sufficiently less than the two times limit such that the heat exchangers are determined to be acceptable for the increased flows.

Based on the revised NSSS parameters for the SGR/Uprating conditions, there is no impact on the auxiliary systems heat exchangers listed in Table 5.9-2. The operating temperature and pressure ranges for these vessels remain bounded by the original design parameters. The original design transients for the auxiliary equipment bound the transients associated with the SGR/Uprating. Therefore, the auxiliary heat exchangers are acceptable for operation at the SGR/Uprating conditions.

5.9.2.3 Auxiliary System Pumps

The NSSS auxiliary pumps that were evaluated are listed in Table 5.9-3. There is no adverse impact on the auxiliary systems pumps listed in Table 5.9-3 as a result of the revised SGR/Uprating conditions. The operating temperature and pressure ranges for these pumps remain bounded by the original design parameters. The original design transients for the auxiliary equipment bound the transients associated with the SGR/Uprating. Therefore, the auxiliary system pumps are acceptable for operation at the SGR/Uprating conditions.

5.9.2.4 Auxiliary System Valves

There is no adverse impact on the auxiliary system valves as a result of the SGR/Uprating. The operating temperature and pressure ranges for the valves remain bounded by the original design parameters. The original design transients for the auxiliary equipment bound the transients associated with the SGR/Uprating. Therefore, the auxiliary system valves are acceptable for operation at the SGR/Uprating conditions.

The CCW system is subject to higher operating temperature as a result of the SGR/Uprating. All of these temperatures remain bounded by the original design conditions. New flow rates also result from the SGR/Uprating. All of these flow rates remain bounded by the original design flow rate. Flow-induced vibration potential is a function of the flow rates (i.e., flow velocities) through the various valves. Therefore, flow-induced vibration is not a concern in any of the auxiliary valves following the SGR/Uprating.

5.9.3 Acceptance Criteria

The maximum system operating temperatures, pressures, and flow rates for the SGR/Uprating program shall be bounded by the original system design conditions for the auxiliary tanks, heat exchangers, pumps and valves.

The original design transients shall bound the revised auxiliary tanks, heat exchangers, pumps and valve transients for the SGR/Uprating program.

5.9.4 Results

A comparison of the revised SGR/Uprating conditions shows that all maximum operating temperatures and pressures for systems within Westinghouse scope are bounded by the existing design basis. Since all tanks, heat exchanger, pumps and valves were designed and manufactured consistent with the system design and applicable codes and standards, all of the NSSS tanks, heat exchangers, pumps and valves are acceptable for the maximum system operating temperatures and pressures resulting from the SGR/Uprating. The auxiliary equipment and NSSS design transients resulting from the SGR/Uprating are bounded by the original HNP design parameters. Therefore, the auxiliary tanks, heat exchangers, pumps and valves remain acceptable for the thermal transients resulting from the SGR/Uprating.

5.9.5 Conclusions

The HNP auxiliary tanks, heat exchangers, pumps and valves are acceptable for the SGR/Uprating conditions, since there is no change to the auxiliary systems operating conditions identified as a consequence of the SGR/Uprating.

The results for the SGR/Uprating are consistent with and continue to comply with the current HNP licensing basis/acceptance requirements. The results obtained with the Δ75 replacement steam generators at the uprated power of 2912.4 MWt bound operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

Table 5.9-1 Harris Nuclear Plant Auxiliary Tanks	
Component	**System**
Recycle Evaporator Reagent Tank	BRS
Recycle Evaporator Condensate Demineralizer	BRS
Recycle Evaporator Feed Demineralizer	CCWS
Component Cooling Surge Tank	CCWS
Boric Acid Batching Tank	CVCS
Chemical Mixing Tank	CVCS
Reactor Coolant Pump Seal Standpipe	CVCS
Volume Control Tank	CVCS
Cation Bed Demineralizer	CVCS
Mixed Bed Demineralizer	CVCS
Pressurizer Relief Tank	RCS
Safety Injection Accumulator Tank	SIS
Boron Injection Tank	SIS
Boron Injection Surge Tank	SIS
Chiller Surge Tank	BTRS
Thermal Regeneration Demineralizer	BTRS
Waste Evaporator Feed Condensate Demineralizer	LWPS
Waste Gas Decay Tank	GWPS
Chemical Drain Tank	LWPS
Waste Evaporator Condensate Tank	LWPS
Reactor Coolant Drain Tank	LWPS
Waste Processing Reagent Tank	LWPS

Table 5.9-2 Harris Nuclear Plant Auxiliary Heat Exchangers	
Component	**System**
Regenerative Hx	CVCS
Residual Hx	RHRS/CCWS
Seal Water Hx	CVCS/CCWS
Excess Letdown Hx	CVCS/CCWS
Letdown Hx	CVCS/CCWS
Reactor Coolant Drain Tank Hx	LWPS/CCWS
Moderating Hx	BTRS
Letdown Chiller Hx	BTRS
Letdown Reheat Hx	BTRS
CCW Hx	CCWS/SWS

Table 5.9-3 Harris Nuclear Plant Auxiliary System Pumps
Boric Acid Transfer Pump
Boron Injection Recirculation Pump
Centrifugal Charging Pump
Chemical Drain Tank Pump
Chiller Pump
Component Cooling Water Pump
Floor Drain Tank Pump
Gas Decay Tank Drain Pump
Hydrotest Pump
Laundry & Hot Shower Tank Pump
RCS Drain Tank Pump
Reactor Makeup Water Pump
Recycle Evaporator Feed Pump
Residual Heat Removal Pump
Spent Resin Sluice Pump
Waste Evaporator Condensate Pump
Waste Evaporator Feed Pump
Waste Gas Compressor Package
Waste Monitor Tank Pump

6.0 NSSS ACCIDENT ANALYSIS

6.0.1 Initial Condition Uncertainties

6.0.1.1 Introduction

This section addresses the initial condition uncertainties used in the accident analyses that were reanalyzed or evaluated to support the Harris Nuclear Plant (HNP) operation at the SGR/Uprating conditions. The uncertainties are included in the non-LOCA analyses, large and small break LOCA, LOCA forces (which are provided as input to component structural analyses), and main steamline break and LOCA mass and energy releases (which are provided as input to the containment integrity analyses).

Six parameters include initial condition steady-state uncertainties that are explicitly modeled in various transient and accident analyses.

- Pressurizer Pressure—Automatic pressurizer pressure control system
- RCS T_{avg} —Automatic reactor control system
- Reactor Power—Daily calorimetric power measurement [Rated Thermal Power (RTP)] used to normalize power range instruments
- RCS Total Flow—Loop RCS flow measurements based on RCS loop flow channels normalized to a once per fuel cycle calorimetric RCS flow measurement to verify Thermal Design Flow (TDF)
- Steam Generator Water Level—Automatic steam generator water level control system
- Pressurizer Water Level—Automatic pressurizer water level control system

The uncertainty calculations were performed for HNP with the plant-specific instrumentation and plant calibration and calorimetric procedures.

6.0.1.2 Description of Analyses and Evaluations

The uncertainty analysis uses the square root of the sum of the squares (SRSS) method to combine the uncertainty components of an instrument channel in an appropriate combination of those components, or groups of components, which are statistically independent. Those uncertainties that are not independent are arithmetically summed to produce groups that are independent of each other, which can then be statistically combined.

6.0.1.3 Acceptance Criteria

The acceptance criteria for the initial condition uncertainties are the values used in the FSAR Chapter 15 analysis. The design verified uncertainties must be less than or equal to the initial condition uncertainty values used in the FSAR Chapter 15 analysis.

6.0.1.4 Results

The results of the uncertainty analysis are provided in Table 6.0-1 and in References 1 and 2.

6.0.1.5 Conclusions

The results are consistent with and continue to comply with the current HNP licensing basis/acceptance requirements. The results obtained with the Δ75 replacement steam generators at the uprated NSSS power of 2912.4 MWt bound operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

6.0.1.6 References

1. WCAP-12340, Rev.1, "Westinghouse Improved Thermal Design Procedure Instrument Uncertainty Methodology for Carolina Power & Light Harris Nuclear Plant (for Uprate to 2912.4 MWt - NSSS Power and Replacement Steam Generators)," January 2000.

2. WCAP-15239, Rev.0, "Westinghouse Pressurizer Water Level Control Uncertainty Methodology for Harris Nuclear Plant (Uprate to 2912.4 MWt - NSSS Power)," November 1999.

Table 6.0-1 Summary of Initial Condition Uncertainties		
Parameter	**Calculated Uncertainty**	**Uncertainty Allowance Used in Safety Analysis**
Pressurizer Pressure	[ ]$^{(a,c)}$ [ ]$^{(a,c)}$	+38.0 psi -50.0 psi
T_{avg}	[ ]$^{(a,c)}$ [ ]$^{(a,c)}$	+6.0°F -6.8°F
Power	[ ]$^{(a,c)}$	±2.0% RTP (random)
RCS Flow (plant computer) (control board indicator)	±1.9% Flow (random) ±2.1% Flow (random)	±2.2% TDF (random) (includes 0.1% bias for feedwater venturi fouling)
Steam Generator Water Level (100% RTP)	[ ]$^{(a,c)}$ [ ]$^{(a,c)}$	Not Applicable
Steam Generator Water Level (0% RTP)	[ ]$^{(a,c)}$ [ ]$^{(a,c)}$	Not Applicable
Pressurizer Water Level (100% RTP)	[ ]$^{(a,c)}$ [ ]$^{(a,c)}$	±6.75% (random)

6.1 Loss-Of-Coolant Accident (LOCA) Transients

6.1.1 Large Break LOCA

This analysis is outside the Westinghouse scope of supply on the SGR/Uprating program. This information will be provided by CP&L or other vendors.

6.1.2 Small Break LOCA

This analysis is outside the Westinghouse scope of supply on the SGR/Uprating program. This information will be provided by CP&L or other vendors.

6.1.3 Post-LOCA Long-Term Core Cooling

6.1.3.1 Introduction

Post-LOCA long-term core cooling (LTCC) analyses are performed to satisfy the requirements of 10 CFR 50.46(b)(5), Long-term cooling. The requirements of 10 CFR 50.46(b)(5) are stated below:

> Long-term cooling. After any calculated successful initial operation of the Emergency Core Cooling System (ECCS), the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

There are three aspects of the post-LOCA LTCC analyses that must be addressed, as follows:

1. Maintaining the core in a subcritical condition following a LOCA.

2. Preventing boron precipitation in the core by calculating a suitable time to switch from cold-leg recirculation to hot-leg recirculation.

3. Establishing the minimum flow requirements for the hot-leg recirculation configuration.

The first item is addressed in this section, and the second and third issues are addressed in Section 6.1.4.

The Westinghouse position on maintaining the core in a sub-critical condition post-LOCA, initially summarized in Reference 1, states that subcriticality for large breaks is maintained without credit for control rods. For the SGR/Uprating program, a range of Mixed Mean Sump Boron Concentrations (MMSBC) corresponding to various initial Reactor Coolant System (RCS) boron concentrations was determined. Note that the calculation of MMSBC is primarily impacted by the change in RCS mass associated with the Model $\Delta 75$ replacement steam generator, as the change in RCS mass associated with uprated power conditions is not considered significant. The minimum boron requirement for post-LOCA sub-criticality will be verified on a cycle-specific basis during the reload process.

For the post-LOCA sump boron calculation, the key inputs are the minimum masses and boron levels of the fluids that have the potential to arrive in the sump, post-LOCA, along with the maximum masses of fluid sources that are potential dilution sources for the sump. The primary boron sources include the Refueling Water Storage Tank (RWST), the RCS, the Accumulators, and various ECCS and Containment Spray piping. Dilution sources include the Containment Spray Additive Tank and associated piping, and the (deactivated) Boron Injection Tank (BIT) and associated piping.

To generate the mixed mean sump boron curve, the pre-trip RCS boron concentration for peak-xenon conditions is assumed to be 100 ppm lower than the equilibrium xenon case. The equation that generates the mixed mean sump boron curve is also expressed so that an equivalent curve can be generated for the highest reasonable postulated peak to equilibrium xenon RCS boron ratio for the particular core design considered.

6.1.3.2 Description of Analyses and Evaluations

The minimum MMSBC is calculated by assuming that all available post-LOCA boron sources to the sump are at minimum volumes, masses, and boron concentrations, while all dilution sources to the sump are at maximum volumes, masses, and minimum (0 ppm) boron concentration. A curve is generated for the sump boron based on the various specified initial RCS boron concentrations. This curve is then used to verify that recriticality cannot occur post-LOCA for the MMSBC associated with the particular core designs used on a cycle-specific basis.

6.1.3.3 Acceptance Criteria

The acceptance criteria for the LTCC post-LOCA sump boron calculation is that the core design must remain such that the minimum post-LOCA sump boron concentration remains sufficient to preclude a return to criticality in the long term post-LOCA.

6.1.3.4 Results

The results of the LTCC hot-leg switchover time calculation are presented below.

Mixed Mean Sump Boron Calculation: The results were tabulated for RCS boron concentration of 0 and 1500 ppm assuming the pre-trip RCS boron concentration for peak xenon concentration to be 100 ppm lower than the equilibrium xenon case. The straight-line curve fit for the two points provided above is:

$$y = 0.20969\,x + 1839.35$$

An alternate equation is provided below where "n" is the difference in boron concentration between the peak and equilibrium xenon concentration case. (See Figure 6.1.3-1.)

$$y = 0.20969\,x + 0.038036\,n + 1835.542$$

6.1.3.5 Conclusions

The Westinghouse licensing position for satisfying the requirements of 10CFR50.46 Paragraph (b) Item (5), "Long-Term Cooling" is documented in Reference 1. The Westinghouse position is that the core will remain subcritical post-LOCA by borated water from various ECCS water sources residing in the RCS and containment sump. Since credit for control rod insertion

is not taken for large-break LOCA, the borated ECCS water provided by the accumulators and RWST must have a sufficiently high boron concentration so that, when mixed with other sources of borated and non-borated water, the core will remain subcritical, assuming that all control rods remain withdrawn from the core.

The revised post-LOCA long-term core cooling boron limit curve is used for cycle-specific core designs to ensure post-LOCA subcriticality for either the current NSSS power of 2787.4 MWt or the uprated power of 2912.4 MWt with the Model Δ75 replacement steam generators. The curve has been generated such that the revised post-LOCA LTCC analyses remain consistent with the current HNP acceptance requirements; the core remains subcritical post-LOCA, and decay heat can be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

6.1.3.6 References

1. Westinghouse Technical Bulletin NSID-TB-86-08, "Post-LOCA Long-Term Cooling: Boron Requirements," October 31, 1986.

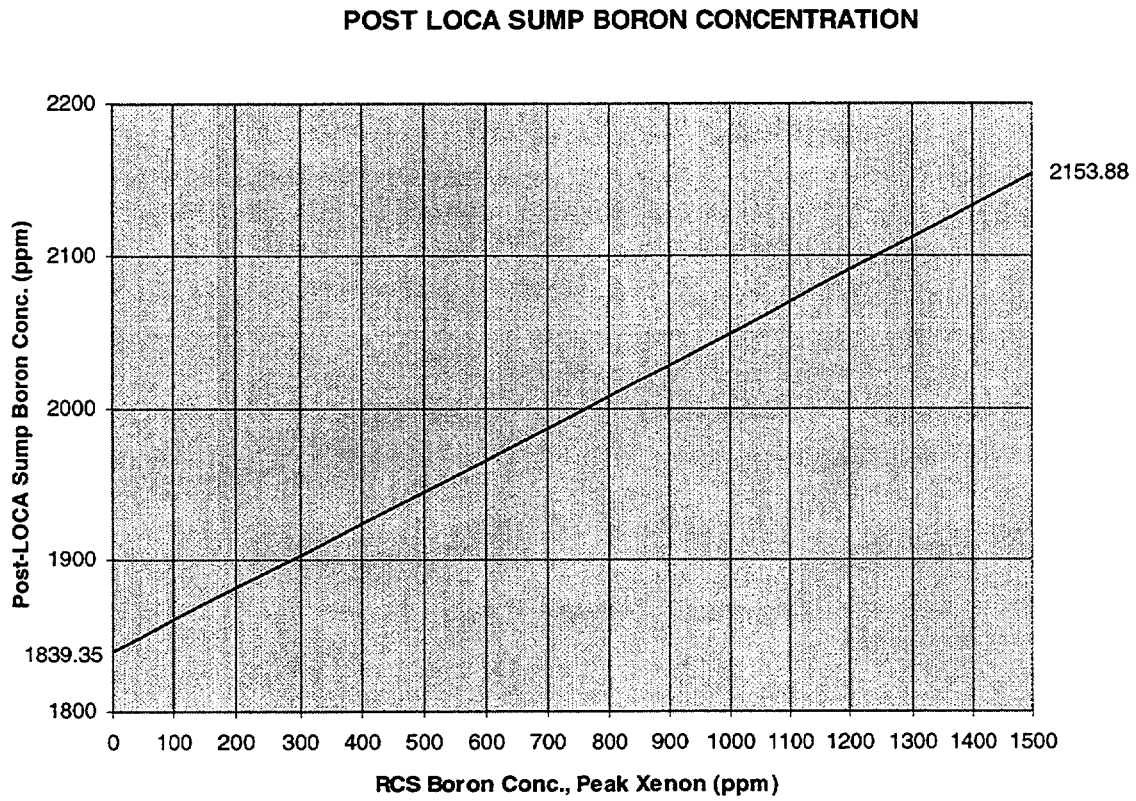

Figure 6.1.3-1 Post-LOCA Sump Boron Concentration

6.1.4 Hot-Leg Switchover

6.1.4.1 Introduction

As described in Section 6.1.3, there are three aspects of the post-LOCA LTCC analyses. The first is maintaining the core in a subcritical condition following a LOCA, which is addressed in Section 6.1.3. The second is preventing boron precipitation in the core by calculating a suitable time to switch from cold-leg recirculation to hot-leg recirculation. The third is establishing the minimum flow requirements for the hot-leg recirculation configuration.

The second issue, preventing boron precipitation in the core through switchover to hot-leg recirculation, is fundamentally described by Reference 1, which is consistent with the current HNP licensing basis as documented in FSAR Section 6.3.2.5.2.3.

The third issue, establishing minimum flow requirements for the hot-leg switchover (HLSO) time for various configurations, is primarily documented in Reference 2 (issued by Westinghouse) and Reference 3 (issued by the NRC). These minimum flow requirements, which depend on calculated core boil-off rates, can then be used to establish the minimum allowable HLSO time based on the decay heat assumptions required in 10CFR50 Appendix K.

6.1.4.2 Description of Analyses and Evaluations

The calculation of the maximum allowable hot-leg switchover time begins with a calculation of the maximum Mixed Mean Sump Boron Concentration (MMSBC). This is calculated assuming that all available post-LOCA boron sources to the sump are at maximum volumes, masses, and boron concentrations, while all dilution sources to the sump are at minimum volumes and masses, and maximum boron concentrations. Then, a minimum vessel mixing volume is computed and a core boil-off calculation is performed, which assumes that fluid enters the vessel at the sump boron concentration, and leaves the vessel as steam (0 ppm boron concentration). The steam is then condensed in containment and added to the sump. As boron accumulates in the vessel mixing volume, boron is depleted from the sump. When the boron concentration in the vessel mixing volume gets to within 4 weight percent of the solubility limit for boron at 14.7 psia (a value of 23.53 weight percent), the maximum allowable hot-leg switchover time is established.

For plants such as the HNP where both high-head and low-head safety injection are aligned to inject to the hot legs at the hot-leg switchover time, a second calculation is performed to establish the required cycling time. This calculation assumes that the initial LOCA occurs in the hot leg, so boron buildup in the vessel does not begin until the initial hot-leg switchover occurs. The second calculation assumes flow comes in from the hot-legs only and similarly is carried out until the 23.53 weight percent boron limit is reached.

Note that for the first of these calculations, it is conservatively assumed that the recirculation flow enters the lower plenum at 212°F (no subcooling). For the second calculation to establish the cycling time, a higher subcooling value (based upon 180°F Residual Heat Removal (RHR) outlet temperatures) is considered justified.

6.1.4.3 Acceptance Criteria

The two acceptance criteria for the maximum allowable hot-leg switchover time calculation are as follows:

- The boron buildup process must be halted by switchover to hot-leg recirculation by the time the margin to the boron solubility limit in the vessel is 4 weight percent (23.53 weight percent boron). Thus, the maximum allowable HLSO time is established by the above described calculation to establish the minimum credible time under which the vessel could reach the boron solubility limit for the range of operating conditions considered.

- This maximum allowable HLSO time must be validated as an acceptable upper limit on the cycling time between hot-leg injection and cold-leg injection, due to HNP's ECCS configuration. Provided the cycling time calculated exceeds the HLSO time calculated, this acceptance criterion is shown to be met.

The two acceptance criteria for the minimum flows following hot-leg switchover are as follows:

- Flow following HLSO must be sufficient to preclude a reduction of core liquid inventory. This ensures compliance with the requirements of 10 CFR 50.46 for the long term.

- Flow following HLSO must be sufficient to preclude precipitation of boron in the vessel (with a 4 weight-percent margin to the solubility limit).

Any time beyond which both of these criteria are met is considered acceptable to enter hot-leg recirculation. With the first criterion met, the core is demonstrated to remain quenched (and potential loss of vessel inventory is precluded) when hot-leg recirculation is established. If the second criterion is not met, the commitment to preclude boron precipitation cannot be demonstrated. Reasonable justifications may be established to support that boron precipitation in and of itself does not constitute a safety issue. Nevertheless both acceptance criteria are considered in this analysis to establish the minimum acceptable HLSO time.

6.1.4.4 Results and Conclusions

The analysis demonstrates that the current HLSO time and cycling time of 6.5 hours remain acceptable. The maximum HLSO time based on an uprated core power of 2900 MWt is 8.5 hours. The current HLSO time of 6.5 hours as discussed in FSAR Section 6.3.2.5.2.3 was assumed to be the initial time for the cycling calculation. Using the alternating hot-leg and cold-

leg recirculation method, the maximum time for cycling between hot-leg and cold-leg recirculation is 14 hours after initially switching over to hot-leg recirculation. The minimum HLSO time based on recirculation flows requirements is 3 hours.

The results obtained from the HLSO calculation for the Model Δ75 replacement steam generators at the uprated NSSS power of 2912.4 MWt bound operation with the Model Δ75 replacement steam generator at the current NSSS power of 2787.4 MWt.

6.1.4.5 References

1. Westinghouse Letter CLC-NS-309, C. L. Caso to T. M. Novak (NRC), April 1, 1975.

2. Westinghouse Nuclear Safety Advisory Letter NSAL-92-010, "Hot Leg Switchover Methodology," January 9, 1993.

3. NRC Information Notice 93-66: "Switchover to Hot-Leg Injection Following a Loss-of-Coolant Accident in Pressurized Water Reactors," August 16, 1993.

6.1.5 Rod Ejection Accident Analysis

This analysis is outside the Westinghouse scope of supply on the SGR/Uprating program. This information will be provided by CP&L or other vendors.

6.2 Non-LOCA Transients

This analysis is outside the Westinghouse scope of supply on the SGR/Uprating program. This information will be provided by CP&L or other vendors.

6.3 Steam Generator Tube Rupture Transient

In support of the Harris Nuclear Plant (HNP) SGR/Uprating program, an evaluation for a design basis steam generator tube rupture (SGTR) event has been performed to demonstrate that the potential consequences are acceptable. The evaluation discussed herein considers operation with a full power average temperature (T_{avg}) of 588.8°F and assumes that up to 10 percent of the steam generator tubes are plugged. The analysis supports a main feedwater temperature window of 375°F to 440°F. Operation with the Model Δ75 replacement steam generators at the current NSSS power level of 2787.4 MWt was also considered.

The major hazard associated with a SGTR event is the radiological consequences resulting from the transfer of radioactive reactor coolant to the secondary side of the ruptured steam generator and subsequent release of radioactivity to the atmosphere. Therefore, an analysis must be performed to assure that the offsite radiation doses resulting from a SGTR are within the allowable guidelines. One of the major concerns for a SGTR is the possibility of steam generator overfill since this could potentially result in a significant increase in the offsite radiation doses. Therefore, an analysis was performed to demonstrate margin to steam generator overfill, assuming the limiting single failure relative to overfill. The analysis confirmed that steam generator overfill does not occur. A thermal and hydraulic analysis was also performed to determine the input for use in calculating the offsite radiation doses, assuming the limiting single failure relative to offsite doses without steam generator overfill. The limiting single failure assumptions for these analyses were evaluated.

Plant response to the event was modeled using the LOFTTR2 computer code with conservative assumptions of break size and location, condenser availability and initial secondary water mass in the ruptured steam generator. The analysis methodology includes the simulation of the operator actions for recovery from a SGTR based on the HNP Emergency Operating Procedures, which are based on the Westinghouse Owners Group Emergency Response Guidelines.

The LOFTTR2 analyses were performed for the time period from the SGTR until the primary and secondary pressures are equalized (break flow termination). In the margin to overfill analysis presented in Section 6.3.1, the water volume in the secondary side of the ruptured steam generator was calculated as a function of time to demonstrate that overfill does not occur. The thermal and hydraulic analysis to develop input to the radiological consequences analysis is presented in Section 6.3.2. In this analysis the primary to secondary break flow and the steam releases to the atmosphere from both the ruptured and intact steam generators were calculated for use in determining the activity released to the atmosphere. The mass releases were calculated with the LOFTTR2 program from the initiation of the event until termination of the break flow. For the time period following break flow termination, steam releases from and feedwater flows to the intact and ruptured steam generators were determined from a mass and energy balance using the calculated reactor coolant system (RCS) and steam generator conditions at the time of leakage termination. The mass release information is used to calculate the radiation doses at the site boundary and low population zone and to the operators in the control room. The radiological consequences analysis is presented in Section 6.3.3.

6.3.1 Analysis of Margin to Steam Generator Overfill

6.3.1.1 Introduction

The SGTR analyses were performed for HNP using the analysis methodology developed in WCAP-10698 (Reference 1) and Supplement 1 to WCAP-10698 (Reference 2). The methodology was developed by the SGTR Subgroup of the Westinghouse Owners Group (WOG) and was approved by the Nuclear Regulatory Commission (NRC) in Safety Evaluation Reports (SERs) dated December 17, 1985 and March 30, 1987. The methodology was developed for use with the LOFTTR2 program, an updated version of the LOFTTR1 program. The LOFTTR1 program was developed as part of the revised SGTR analysis methodology and was used for the SGTR evaluations in References 1 and 2. This is the same methodology employed in the most recent analyses performed by Westinghouse for HNP, documented in WCAP-12403 and Supplement 1 to WCAP-12403 (References 3 and 4).

An analysis was performed to determine the margin to steam generator overfill for a design basis SGTR event for HNP. The analysis was performed using the LOFTTR2 program and the methodology developed in Reference 1, and using the plant specific parameters for HNP. This section includes the methods and assumptions used to analyze the SGTR event, as well as the sequence of events for the recovery and the calculated results.

6.3.1.2 Description of Analyses and Evaluations

The margin to overfill analysis assumes that the plant is operating with the feedwater temperature at the low end of the temperature window, since this results in a higher mass of water in the steam generator at the start of the event, which limits the amount of break flow and auxiliary feedwater (AFW) that can accumulate in the ruptured steam generator without forcing water into the steamlines. Maximum (10-percent) tube plugging is assumed in the thermal-hydraulic analysis to determine the margin to overfill since this reduces the heat transfer to the ruptured steam generator, minimizing the amount of mass released from the steam generator due to steaming, which in turn reduces the margin to overfill. The reduced heat transfer also prolongs the cooldown period, leading to delayed break flow termination. Although maximum tube plugging results in a lower initial water mass in the ruptured steam generator, which increases the available margin to overfill at the start of the event, this is not more limiting than the lower heat transfer effects described above. This has been confirmed via sensitivity runs specifically made for the HNP SGR/Uprating program.

Design Basis Accident

The accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet (cold leg) side of the steam generator. The location of the break on the cold side of the steam generator results in higher primary to secondary leakage than a break on the hot side of the steam generator as determined by Reference 1. It was also assumed that a

loss of offsite power occurs at the time of reactor trip, and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip.

The potentially limiting single failures with respect to margin to steam generator overfill for a SGTR from Reference 1 are outlined below.

1. Auxiliary Feedwater (AFW) Flow Control Valve Failure

   The AFW control valves are normally open and are used to control inventory in the intact steam generators and terminate feedwater flow to the ruptured steam generator. A failure of the ruptured steam generator control valve would require the operator to perform additional action to stop the associated AFW pump in order to terminate AFW flow to the ruptured steam generator. It is assumed that 120 seconds (2 minutes) of operator action time will be required to terminate AFW flow to the ruptured steam generator by stopping the associated AFW pump. This 2 minutes is added to the time for AFW isolation assumed without this failure, which is discussed later in this section and identified in Table 6.3.1-1. The additional AFW provided to the ruptured steam generator during that 2 minutes decreases the margin to overfill. The continued AFW flow also results in a reduction in steaming from the ruptured steam generator, which reduces the margin to overfill.

2. Intact Steam Generator Power Operated Relief Valve (PORV) Failure

   Since offsite power is assumed to be lost at reactor trip for the SGTR analysis, the PORVs are relied upon to cool the reactor coolant system. Failure of a PORV on an intact steam generator to open on demand reduces the steam releases capability to that provided by a single PORV. This increases the time required for the cooldown, resulting in increased break flow and a reduction in the margin to steam generator overfill. As indicated in Reference 1 this is typically the limiting single failure for a three loop plant.

   The analyses performed for the HNP replacement steam generator and uprate programs specifically evaluated these two potentially limiting single failures and determined that the limiting failure for the margin to overfill analysis is the failure of the PORV on one of the two intact steam generators to open for the cooldown. The analysis presented in this report models this failure.

Conservative Assumptions

Plant responses until break flow termination were calculated using the LOFTTR2 computer code. The conservative conditions and assumptions which were used in Reference 1 were also used in the LOFTTR2 analysis to determine margin to steam generator overfill for HNP with the exception of the following differences.

1. Reactor Trip and Turbine Runback

   A turbine runback can either be initiated automatically or the operator can manually reduce the turbine load to attempt to prevent a reactor trip on overtemperature-ΔT. Although turbine runback is simulated in this analysis, credit is not taken for delaying reactor trip. Until reactor trip and the assumed loss of offsite power, the main feedwater control system is assumed to maintain a constant steam generator water level. Therefore, until reactor trip, the break flow does not reduce the margin to overfill. An earlier reactor trip will result in an earlier increase in the ruptured steam generator water volume and earlier initiation of AFW. These effects will result in an increased secondary mass in the ruptured steam generator at the time of isolation since the isolation is assumed to occur at a fixed time after the SGTR occurs rather than at a fixed time after reactor trip. For this analysis the time of reactor trip on overtemperature-ΔT was determined by modeling the HNP protection system to occur at approximately 113 seconds. The effect of turbine runback was simulated until reactor trip at the rate of 10 percent per minute. The effect of turbine runback was conservatively simulated by increasing the secondary mass by the differential between 81 percent and 100 percent power and performing the analysis at 100 percent power.

2. Steam Generator Secondary Mass

   A higher initial secondary water mass in the ruptured steam generator was determined by Reference 1 to be conservative for overfill. As noted above, turbine runback was assumed to be initiated and was simulated by artificially increasing the initial steam generator water mass. The initial steam generator total fluid mass was assumed to be 10 percent above the nominal full power fluid mass, plus the differential mass between 100 percent power and 81 percent power to simulate the effect of turbine runback.

3. AFW System Operation

   For this analysis the maximum AFW flow rate of 500 gpm to the ruptured steam generator was assumed to be initiated immediately after reactor trip with no startup delays credited.

Operator Action Times

In the event of a SGTR, the operator is required to take actions to stabilize the plant and terminate the primary to secondary leakage. The operator actions for SGTR recovery are

provided in the HNP EOPs PATH-2, and major actions were explicitly modeled in this analysis. The operator actions modeled include identification and isolation of the ruptured steam generator, cooldown and depressurization of the RCS to restore inventory and termination of SI to stop primary to secondary leakage. These operator actions are described below.

1. Identify the ruptured steam generator.

   High secondary side activity, as indicated by the condenser vacuum pump effluent radiation monitor, steam generator blowdown line radiation monitor, or main steamline radiation monitor, typically will provide the first indication of a SGTR event. The ruptured steam generator can be identified by a mismatch between steam and feedwater flow, high activity in a steam generator water sample, or a high radiation indication on the corresponding main steamline radiation monitor. For a SGTR that results in a reactor trip at high power as assumed in this analysis, the steam generator water level as indicated on the narrow range will decrease significantly for all of the steam generators. The AFW flow will begin to refill the steam generators, distributing approximately equal flow to each of the steam generators. Since primary to secondary leakage adds additional inventory to the ruptured steam generator, the water level will increase more rapidly in that steam generator. This response, as displayed by the steam generator water level instrumentation, provides confirmation of a SGTR event and also identifies the ruptured steam generator.

2. Isolate the ruptured steam generator from the intact steam generators and isolate feedwater to the ruptured steam generator.

   Once the steam generator with a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator. In addition to minimizing radiological releases, this also reduces the possibility of filling the ruptured steam generator by (1) minimizing the accumulation of feedwater flow and (2) enabling the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary to secondary leakage. In the HNP EOP for steam generator tube rupture, the operator is directed to maintain the level in the ruptured steam generator between 10 percent and 50 percent on the narrow range instrument. To model the isolation time using the methodology in Reference 1, it was assumed that AFW flow to the ruptured steam generator would be isolated when level in the steam generator reached 30 percent narrow range level or at 10 minutes, whichever is longer. Complete isolation of steam flow from the ruptured steam generator is verified when the narrow range level reaches 30 percent on the ruptured steam generator or at 12 minutes after initiation of the SGTR, whichever is longer.

3. Cooldown the Reactor Coolant System (RCS) using the intact steam generator.

   After isolation of the ruptured steam generator, the RCS is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured steam generator pressure

by dumping steam from only the intact steam generators. This ensures adequate subcooling in the RCS after depressurization to the ruptured steam generator pressure in subsequent actions. If offsite power is available, the normal steam dump system to the condenser can be used to perform this cooldown. However, if offsite power is lost, the RCS is cooled using the power-operated relief valves (PORVs) on the intact steam generators. Since offsite power is assumed to be lost at reactor trip for this analysis, the cooldown was performed by dumping steam via the PORVs on the intact steam generators. Due to the single failure assumed in this analysis only one intact steam generator is used for the cooldown.

4. Depressurize the RCS to restore reactor coolant inventory.

   When the cooldown is completed, SI flow will tend to increase RCS pressure until break flow matches SI flow. Consequently, SI flow must be terminated to stop primary to secondary leakage. However, adequate reactor coolant inventory must first be assured. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after SI flow is stopped. Since leakage from the primary side will continue after SI flow is stopped until RCS and ruptured steam generator pressures equalize, an "excess" amount of inventory is needed to ensure the pressurizer level remains on span. The "excess" amount required depends on the RCS pressure and reduces to zero when the RCS pressure equals the pressure in the ruptured steam generator.

   The RCS depressurization is performed using normal pressurizer spray if the reactor coolant pumps (RCPs) are running. Since offsite power is assumed to be lost at the time of reactor trip, the RCPs are not running and thus normal pressurizer spray is not available. Therefore, the depressurization is modeled using a pressurizer power operated relief valve (PORV).

5. Terminate SI to stop primary to secondary leakage.

   The previous actions will have established adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that the SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to terminate primary to secondary leakage. Primary to secondary leakage will continue after the SI flow is stopped until the RCS and ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent re-pressurization of the RCS and re-initiation of leakage into the ruptured steam generator.

Since these major recovery actions will be modeled in the SGTR analysis, it is necessary to establish the times required to perform these actions. Although the intermediate steps between the major actions will not be explicitly modeled, it is also necessary to account for the time required to perform the steps. It is noted that the total time required to complete the recovery

operations consists of both operator action time and system, or plant, response time. For instance, the time for each of the major recovery operations (i.e., RCS cooldown) is primarily due to the time required for the system response, whereas the operator action time is reflected by the time required for the operator to perform the intermediate action steps.

The operator action times to identify and isolate the ruptured steam generator, to initiate RCS cooldown, to initiate RCS depressurization, and to perform safety injection termination were developed for the design basis analysis in Reference 1. CP&L has determined the corresponding operator action times to perform these operations for HNP. The operator actions and the corresponding operator action times used for the HNP analysis are listed in Table 6.3.1-1.

These operator action times are different from those modeled in the Reference 3 and 4 analyses. For the replacement steam generator analysis, the operator actions for isolation of AFW flow to the ruptured steam generator and isolation of steam flow from the ruptured steam generator have been separated, while in the Reference 3 and 4 analyses, these actions were performed simultaneously. Both isolation times are earlier than assumed in References 3 and 4. The times from steamline isolation until the cooldown is initiated and from the end of cooldown until the depressurization is initiated have been changed from those assumed in References 3 and 4. The Reference 3 and 4 times are provided in Table 6.3.1-1.

6.3.1.3 Description of Analysis Cases

The LOFTTR2 analysis results for the HNP margin to overfill analysis with Model Δ75 replacement steam generators and operation at the uprated NSSS power of 2912.4 MWt are described below. The sequence of events for this transient is presented in Table 6.3.1-2

Following the tube rupture, reactor coolant flows from the primary into the secondary side of the ruptured steam generator since the primary pressure is greater than the steam generator pressure. In response to this loss of reactor coolant, pressurizer level decreases as shown in Figure 6.3.1-1. The RCS pressure also decreases as shown in Figure 6.3.1-2 as the steam bubble in the pressurizer expands. As the RCS pressure decreases due to the continued primary to secondary leakage, automatic reactor trip occurs on an overtemperature-ΔT trip signal at approximately 113 seconds.

After reactor trip, core power rapidly decreases to decay heat levels. The turbine stop valves close and steam flow to the turbine is terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remain closed due to the loss of condenser vacuum resulting from the assumed loss of offsite power at the time of reactor trip. Thus, the energy transfer from the primary system causes the secondary side pressure to increase rapidly after reactor trip until the steam generator PORVs (and safety valves if their setpoints are reached) lift to dissipate the energy, as shown in Figure 6.3.1-3. As a result of the assumed loss of offsite power, main feedwater flow was assumed to be terminated and AFW flow was assumed to be automatically initiated following

reactor trip. The air supply to the MSIVs is assumed to fail on the loss of offsite power and the MSIVs will go to the closed position as the instrument air supply degrades.

The RCS pressure and pressurizer level continue to decrease after reactor trip as energy transfer to the secondary shrinks the reactor coolant and the tube rupture break flow continues to deplete primary inventory. The decrease in RCS inventory results in a low pressurizer pressure SI signal at approximately 183 seconds. The SI flow increases the reactor coolant inventory and the RCS pressure trends toward the equilibrium value where the SI flow rate equals the break flow rate.

Since offsite power is assumed lost at reactor trip, the RCPs trip and a gradual transition to natural circulation flow occurs. Immediately following reactor trip the temperature differential across the core decreases as core power decays (see Figure 6.3.1-4); however, the temperature differential subsequently increases as the reactor coolant pumps coast down and natural circulation flow develops. The cold-leg temperature trends toward the steam generator temperature as the fluid residence time in the tube region increases. The RCS temperatures continue to slowly decrease due to the continued addition of the auxiliary feedwater to the steam generators until operator actions are initiated to cool down the RCS.

Major Operator Actions

1. Identify and Isolate the ruptured steam generator.

   Recovery actions begin by throttling the auxiliary feedwater flow to the ruptured steam generator and isolating steam flow from the ruptured steam generator. As indicated previously, auxiliary feedwater flow to the ruptured steam generator is assumed to be identified and isolated when the narrow range level reaches 30 percent on the ruptured steam generator or at 10 minutes after initiation of the SGTR, whichever is longer. For the HNP analysis the time to reach 30 percent is less than 10 minutes, and thus the ruptured steam generator is assumed to be isolated at 10 minutes. Also, as indicated previously, complete isolation of steam flow from the ruptured steam generator is verified when the narrow range level reaches 30 percent on the ruptured steam generator or at 12 minutes after initiation of the SGTR, whichever is longer. For the HNP analysis the time to reach 30 percent is less than 12 minutes, and thus the ruptured steam generator is assumed to be isolated at 12 minutes.

2. Cooldown the RCS to establish subcooling margin.

   After isolation of the ruptured steam generator, a 5-minute operator action time is imposed prior to initiating the cooldown. After this time, actions are taken to cool the RCS as rapidly as possible by dumping steam from the intact steam generators. Since offsite power is lost, the RCS is cooled by dumping steam to the atmosphere using the PORV on both intact steam generators. However, as noted previously, the limiting single failure was assumed to be the failure of one of the intact steam generator PORVs to lift on demand. It

was therefore assumed only one of the intact steam generator's PORV is opened at 17 minutes for the RCS cooldown. The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance of 20°F for subcooling uncertainty. When these conditions are satisfied at 1960 seconds, it is assumed that the operator closes the intact steam generator's PORV to terminate the cooldown. This cooldown ensures that there will be adequate subcooling in the RCS after the subsequent depressurization of the RCS to the ruptured steam generator pressure. The reduction in the intact steam generator pressure required to accomplish the cooldown is shown in Figure 6.3.1-3, and the effect of the cooldown on the RCS temperature is also shown in Figure 6.3.1-4. The pressurizer level and RCS pressure also decreases during this cooldown process due to shrinkage of the reactor coolant as shown in Figure 6.3.1-1.

3. Depressurize the RCS to restore inventory.

The RCS depressurization is performed to assure adequate coolant inventory prior to terminating SI flow. A 4-minute operator action time is included prior to the RCS depressurization. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS is depressurized by opening a pressurizer PORV. The RCS depressurization is initiated at 2200 seconds and continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than the allowance of 10 percent for pressurizer level uncertainty, or pressurizer level is greater than 75 percent, or RCS subcooling is less than the 20°F allowance for subcooling uncertainty. For this case, the RCS depressurization is terminated at 2332 seconds because the RCS pressure is reduced to less than the ruptured steam generator pressure and the pressurizer level is above 10 percent. The RCS depressurization (Figure 6.3.1-2) reduces the break flow as shown in Figure 6.3.1-5 and increases SI flow to refill the pressurizer, as shown in Figure 6.3.1-1.

4. Terminate SI to stop primary to secondary leakage.

The previous actions establish adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent re-pressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated at this time if RCS subcooling is greater than the 20°F allowance for subcooling uncertainty, minimum AFW flow is available or at least one intact steam generator level is in the narrow range, the RCS pressure is stable or increasing, and the pressurizer level is greater than the 10-percent allowance for uncertainty.

After depressurization is completed, an operator action time of 3 minutes was assumed prior to SI termination. Since the above requirements are satisfied, SI termination actions were performed at 2512 seconds by closing off the SI flow path. After SI termination the RCS pressure begins to decrease as shown in Figure 6.3.1-2.

The intact steam generator's PORV, which was used for the cooldown, also automatically open (at about 2550 seconds) to dump steam to maintain the prescribed RCS temperature to ensure that subcooling is maintained. When the PORV is opened, the increased energy transfer from primary to secondary also aids in the depressurization of the RCS to the ruptured steam generator pressure. The primary to secondary leakage continues after the SI flow is terminated until the RCS and ruptured steam generator pressures equalize.

6.3.1.4 Acceptance Criteria

The analysis is performed to demonstrate that the secondary side of the ruptured steam generator does not completely fill with water. The available secondary side volume of a single HNP Model Δ75 replacement steam generator is 5545 ft^3. Margin to overfill is demonstrated provided the transient calculated steam generator secondary side water volume is less than 5545 ft^3.

6.3.1.5 Results

The primary to secondary break flow rate throughout the recovery operations is presented in Figure 6.3.1-5. The water volume in the ruptured steam generator is presented as a function of time in Figure 6.3.1-6. The secondary side volume of a single HNP Model Δ75 replacement steam generator, up to the outlet nozzle, is 5545 ft^3. The peak ruptured steam generator water volume of 5285 ft^3 is indicated in Figure 6.3.1-6 showing that there is 260 ft^3 of margin to overfill. No credit is taken for the volume of the nozzle or any steam piping. Therefore, it is concluded that overfill of the ruptured steam generator will not occur for a design basis SGTR for HNP with the replacement steam generators and operation at the uprated NSSS power of 2912.4 MWt.

The difference in plant operating parameters and initial conditions between the uprated and current NSSS power levels has a small impact on the margin to overfill analysis. This has been confirmed via sensitivity runs made specifically for the HNP SGR/Uprating program. The analysis performed at the uprated power level demonstrated significant margin to overfill. Therefore, it is concluded that overfill of the ruptured steam generator will not occur for a design basis SGTR for HNP with the replacement steam generators and operation at the current NSSS power of 2787.4 MWt.

6.3.1.6 Conclusions

It is calculated that overfill of the ruptured steam generator will not occur for a design basis SGTR for HNP with the replacement steam generators and operation at either the uprated (2912.4 MWt) or the current (2787.4 MWt) NSSS power.

6.3.1.7 References

1. WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," Lewis, Huang, Behnke, Fittante, Gelman, August 1987.

2. Supplement 1 to WCAP-10698-P-A, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," Lewis, Huang, Rubin, March 1986.

3. WCAP-12403, "LOFTTR2 Analysis for a Steam Generator Tube Rupture with Revised Operator Action Times for Shearon Harris Nuclear Power Plant," Huang, Lewis, Marmo, Rubin, November 1989.

4. Supplement 1 to WCAP-12403, "Steam Generator Tube Rupture Analysis for Shearon Harris Nuclear Power Plant," Lewis, Lowe, Monahan, Rubin, Tanz , November 1992.

Table 6.3.1-1 Operator Action Times For Design Basis SGTR Analysis		
Action	**Time**	**Reference 3 and 4 Time**
Isolate auxiliary feedwater flow to ruptured SG	Maximum of 10 min. or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG	Maximum of 13 min. 35 sec. or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG
Isolate steam flow from ruptured SG	Maximum of 12 min. or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG	Maximum of 13 min. 35 sec. or LOFTTR2 calculated time to reach 30% narrow range level in the ruptured SG
Operator action time to initiate cooldown	5 min. from time of steamline isolation*	8 min. 1 sec. from time of steamline isolation
Cooldown	Calculated by LOFTTR2	Calculated by LOFTTR2
Operator action time to initiate depressurization	4 minutes from end of cooldown	2 min. 16 sec. from end of cooldown
Depressurization	Calculated by LOFTTR2	Calculated by LOFTTR2
Operator action time to terminate SI following depressurization	Maximum of 3 minutes from end of depressurization or time to satisfy termination criteria	Maximum of 3 minutes from end of depressurization or time to satisfy termination criteria
Pressure equalization	Calculated by LOFTTR2	Calculated by LOFTTR2

* The analysis considered the possibility that the MSIVs would close as a result of the loss of offsite power assumed to occur coincident with reactor trip by isolating the steam lines at that time. The steamline isolation step was retained to model the time when the operators reach the step in the EOPs requiring that isolation of all flow in and out of the ruptured steam generator is isolated.

Table 6.3.1-2 Sequence of Events for Margin to Overfill Analysis	
	Operation at 2912.4 MWt
Event	**Time (seconds)**
Steam Generator Tube Rupture	0
Reactor Trip	113
SI Actuation	183
AFW Flow to Ruptured SG Isolated	600
Ruptured SG Isolated	720
RCS Cooldown Initiated	1020
RCS Cooldown Terminated	1960
RCS Depressurization Initiated	2200
RCS Depressurization Terminated	2332
SI Terminated	2512
Break Flow Terminated	3024

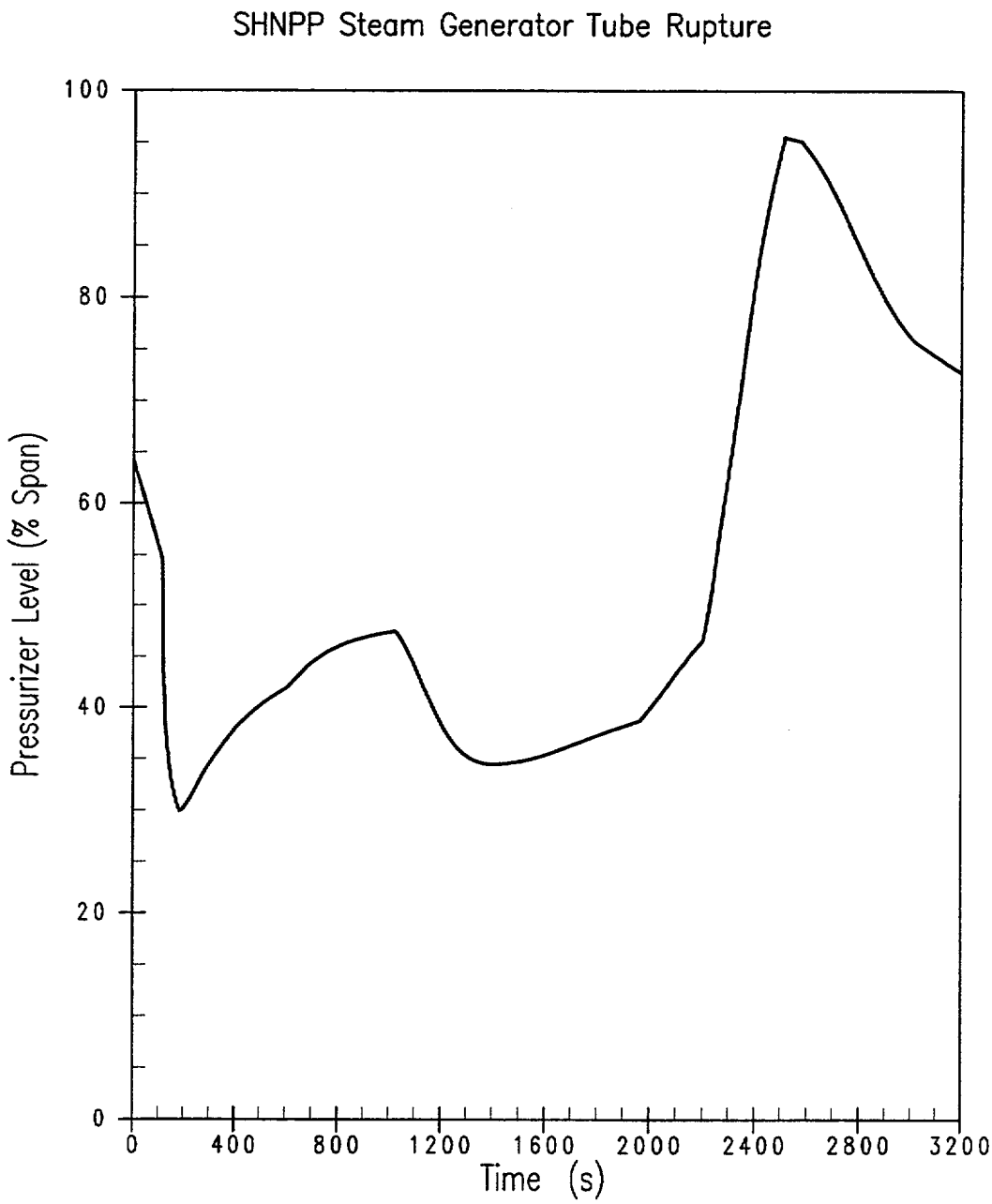

Figure 6.3.1-1
Pressurizer Level –
Margin to Overfill Analysis (2912.4 MWt)

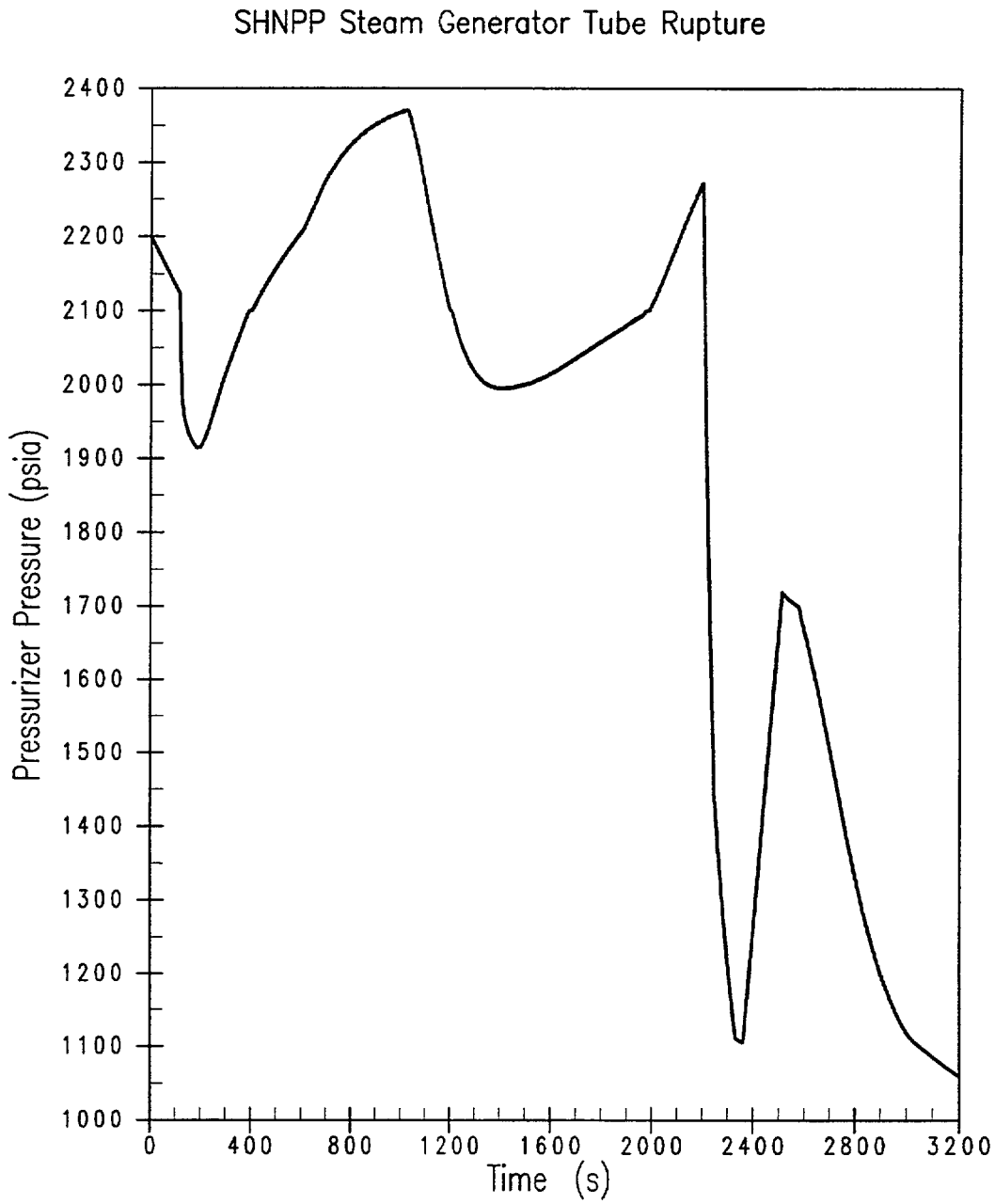

Figure 6.3.1-2
Pressurizer Pressure –
Margin to Overfill Analysis (2912.4 MWt)

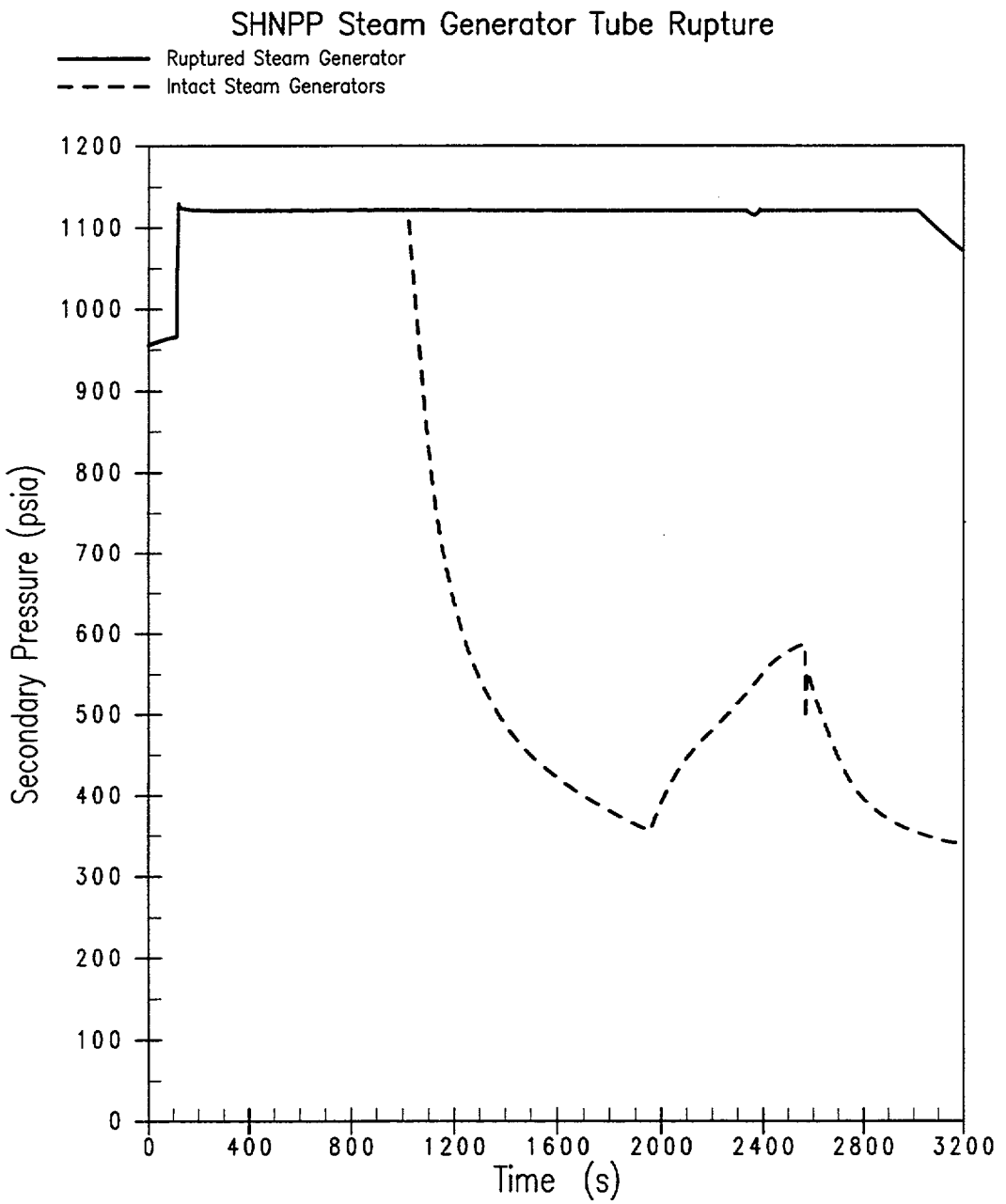

Figure 6.3.1-3
Secondary Pressure –
Margin to Overfill Analysis (2912.4 MWt)

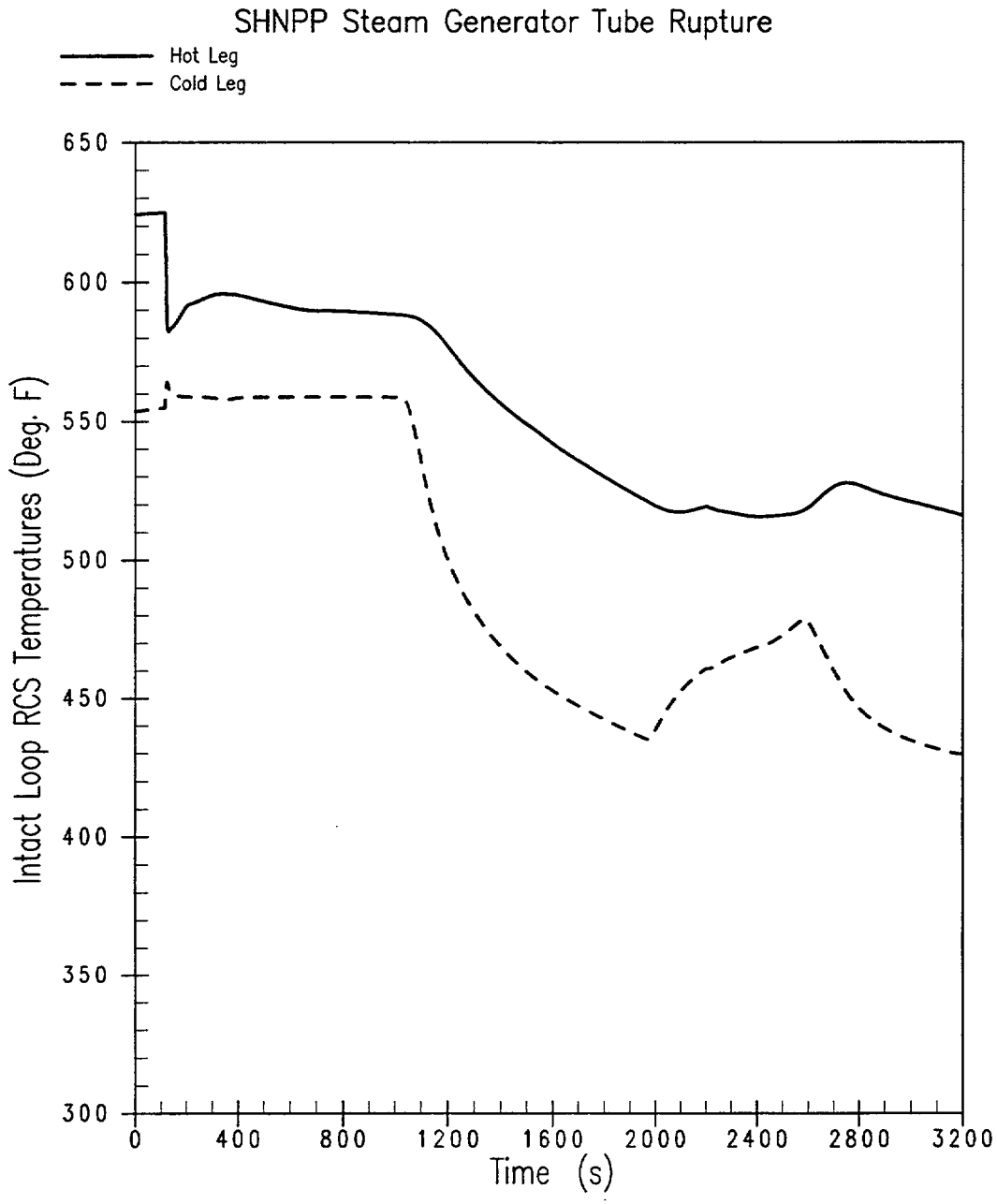

Figure 6.3.1-4
Intact Loop Hot & Cold Leg Temperatures –
Margin to Overfill Analysis (2912.4 MWt)

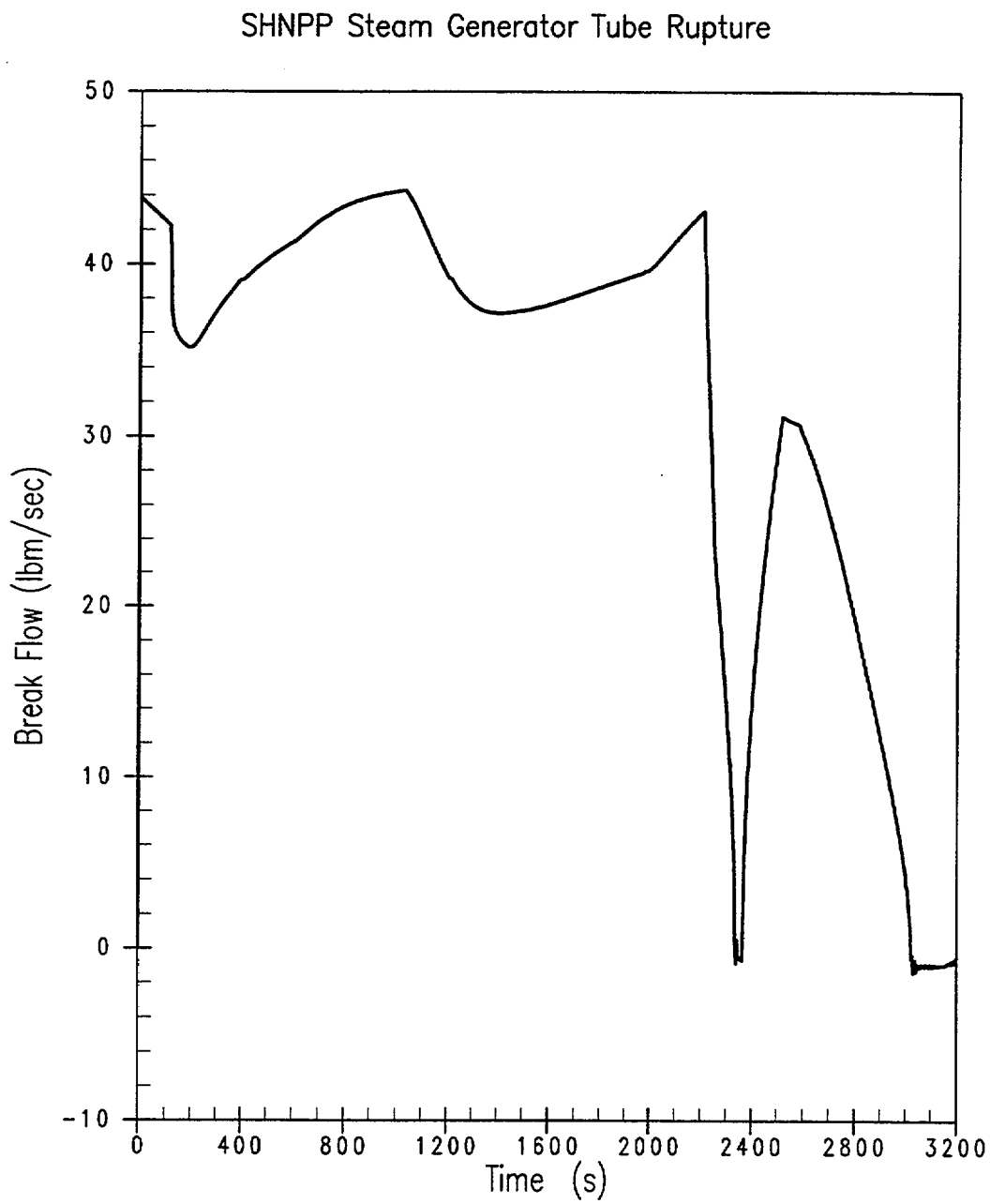

Figure 6.3.1-5
Primary to Secondary Break Flow –
Margin to Overfill Analysis (2912.4 MWt)

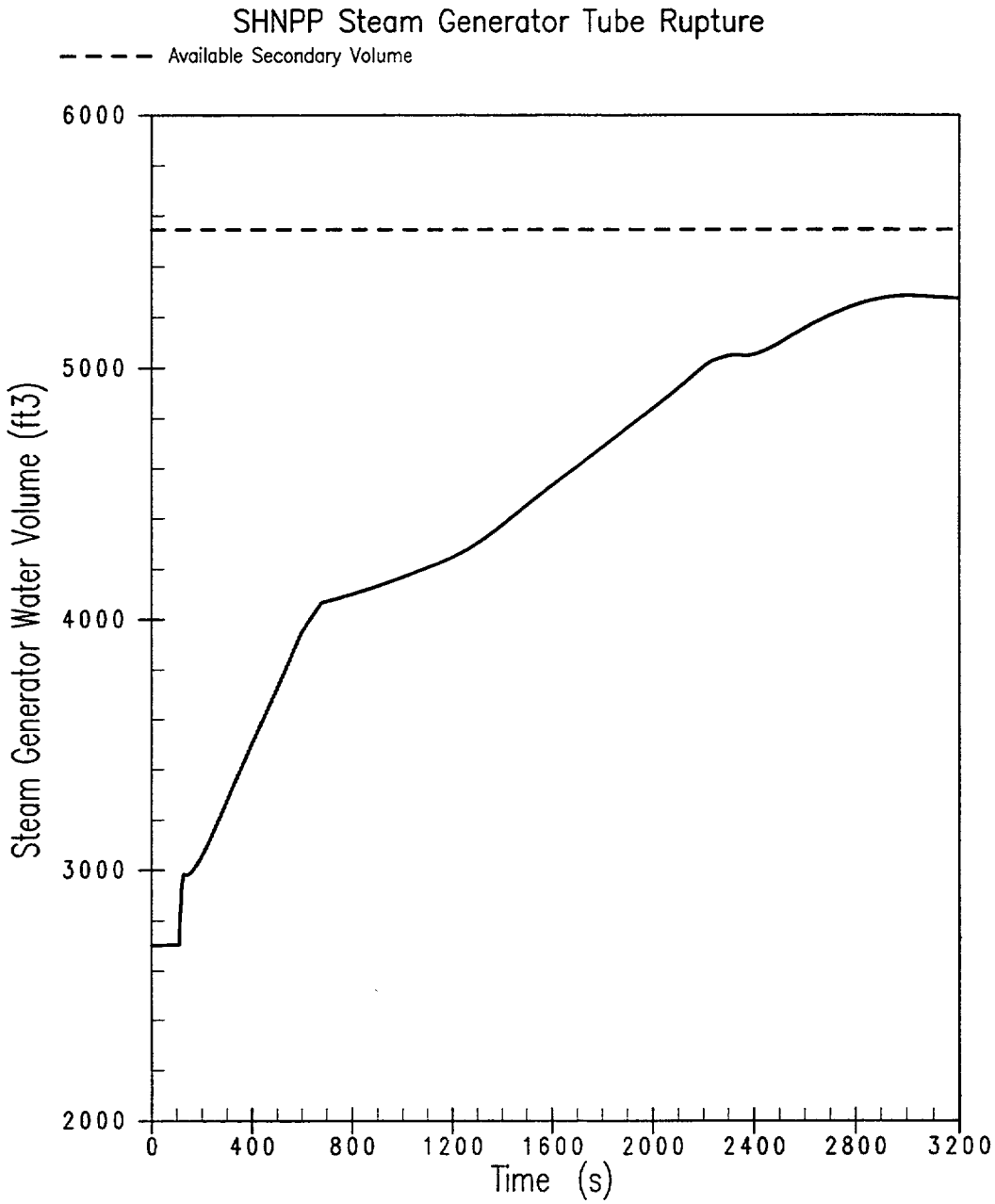

Figure 6.3.1-6
Ruptured SG Water Volume –
Margin to Overfill Analysis (2912.4 MWt)

6.3.2 Thermal and Hydraulic Analysis for Offsite Radiological Consequences

6.3.2.1 Introduction

The Steam Generator Tube Rupture (SGTR) analyses were performed for HNP using the analysis methodology developed in WCAP-10698 (Reference 1) and Supplement 1 to WCAP-10698 (Reference 2). The methodology was developed by the SGTR Subgroup of the Westinghouse Owners Group (WOG) and was approved by the Nuclear Regulatory Commission (NRC) in Safety Evaluation Reports (SERs) dated December 17, 1985 and March 30, 1987. The methodology was developed for use with the LOFTTR2 program, an updated version of the LOFTTR1 program. The LOFTTR1 program was developed as part of the revised SGTR analysis methodology and was used for the SGTR evaluations in References 1 and 2. This is the same methodology employed in the most recent analyses performed by Westinghouse for HNP, documented in WCAP-12403 and Supplement 1 to WCAP-12403 (References 3 and 4).

A thermal and hydraulic analysis was performed to determine the input for the offsite radiological consequences analysis for a design basis SGTR event for HNP. The thermal and hydraulic analysis was performed using the LOFTTR2 program and the methodology developed in References 1 and 2, and using the plant specific parameters for HNP. This section includes the methods and assumptions used to analyze the SGTR event, as well as the sequence of events for the recovery and the calculated results.

6.3.2.2 Description of Analyses and Evaluations

The thermal-hydraulic analysis, which determines the offsite dose mass releases, assumes that the plant is operating with the feedwater temperature at the lower end of the temperature window, since this was determined to result in slightly higher releases. No tube plugging is assumed in the analysis as this maximizes heat transfer to the ruptured steam generator. A high heat transfer rate during the transient maximizes the amount of mass released from the steam generator due to steaming. Although maximum tube plugging results in a lower initial water mass in the ruptured steam generator, which leads to increased steam releases, this is not more limiting than the higher heat transfer corresponding to no tube plugging. This has been confirmed via sensitivity runs specifically made for the HNP replacement steam generator and uprate program.

Design Basis Accident

The design basis accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet (cold-leg) side of the steam generator. The location of the break on the cold side of the steam generator results in higher primary to secondary leakage than a break on the hot side of the steam generator, as determined by Reference 1. However, as indicated subsequently, the break flow flashing fraction was conservatively calculated assuming that all of the break flow comes from the hot-leg side of the steam generator. The combination of these conservative assumptions regarding the break location results in a very conservative

calculation of the offsite radiation doses. It was also assumed that loss-of-offsite power occurs at the time of reactor trip, and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip. Due to the assumed loss of offsite power, the condenser is not available for steam releases once the reactor is tripped. Consequently, after reactor trip, steam is released to the atmosphere through the steam generator PORVs. After reactor trip and loss of offsite power, the RCPs begin to coast down.

Based on the information in Reference 2, the most limiting single failure with respect to offsite doses is a failed open power-operated relief valve (PORV) on the steam generator with the ruptured tube. Failure of this PORV will cause an uncontrolled depressurization of the steam generator, which will increase primary to secondary leakage and the mass release to the atmosphere. Pressure in the ruptured steam generator will remain below that in the primary system until the failed PORV can be isolated, and recovery actions completed.

Conservative Assumptions

The integrated primary-to-secondary break flow and the mass releases from the ruptured and intact steam generators to the condenser and to the atmosphere until break flow termination were calculated with the LOFTTR2 program. This is used in calculating the offsite radiation doses. This section includes a discussion of the methods and assumptions used to analyze the SGTR event and to calculate the mass releases, the sequence of events during the recovery operations, and the calculated results.

Most of the conservative conditions and assumptions used for the margin to overfill analysis are also conservative for the offsite dose analysis, and thus most of the same assumptions were used for both analyses. The major differences in the assumptions that were used for the LOFTTR2 analysis for offsite doses are discussed below.

1. Reactor Trip and Turbine Runback

   An earlier reactor trip is conservative for the offsite dose analysis. Due to the assumed loss of offsite power, the condenser is not available for steam releases once the reactor is tripped. Consequently, after reactor trip, steam is released to the atmosphere through the steam generator PORVs. Thus an earlier trip time leads to more steam released to the atmosphere from the ruptured and intact steam generators. The time of reactor trip was calculated by modeling the HNP protection system and this time was used in the analysis. However, turbine runback was not simulated for the analysis since the increased secondary water mass due to turbine runback is not conservative for the offsite dose analysis.

2. Steam Generator Secondary Mass

A lower initial mass in the ruptured steam generator results in a conservative prediction of offsite doses. The initial steam generator total fluid mass was assumed to be 10-percent below the nominal full-power fluid mass.

3. Auxiliary Feedwater (AFW) System Operation

For this analysis, the minimum AFW flow rate of 390 gpm to the ruptured steam generator was assumed to be initiated 61.5 seconds after reactor trip. A minimum AFW flow rate maximizes steam releases to the atmosphere.

4. Flashing Fraction

When calculating the fraction of break flow that flashes to steam, 100 percent of the break flow is assumed to come from the hot-leg side of the break. Since the tube rupture flow actually consists of flow from the hot-leg and cold-leg sides of the steam generator, the temperature of the combined flow will be less than the hot-leg temperature and the flashing fraction will be correspondingly lower. Thus the assumption is conservative for a SGTR analysis.

Operator Action Times

The major operator actions required for the recovery from a SGTR are discussed in Section 6.3.1.2, and the operator action times used for the margin to overfill analysis are presented in Table 6.3.1-1. The operator action times assumed for the margin to overfill analysis were also used for the offsite dose analysis. However, for the offsite doses analysis, the PORV on the ruptured steam generator was assumed to fail open at the time the ruptured steam generator is isolated. Before proceeding with the recovery operations, the failed-open PORV on the ruptured steam generator is assumed to be isolated by locally closing the associated block valve. CP&L has determined that an operator can locally close the block valve for the PORV on the ruptured steam generator within 20 minutes after the failure. Thus, it was assumed that the ruptured steam generator PORV is isolated at 20 minutes after the valve is assumed to fail open. The operator action time to close the block valve for the PORV on the ruptured steam generator is the same as that modeled in the Reference 3 and 4 analyses. After the ruptured steam generator PORV is isolated, the additional delay time of 5 minutes (Table 6.3.1-1) was assumed for the operator action time to initiate the RCS cooldown.

Mass Releases

The mass releases were determined for use in evaluating the offsite and control room radiological consequences of the SGTR using the methodology of Reference 2. The steam releases from the ruptured and intact steam generators, the feedwater flow to the ruptured and intact steam

generators, and primary to secondary break flow into the ruptured steam generator were determined for the period from accident initiation until 2 hours after the accident and from 2 to 8 hours after the accident. The releases for 0 to 2 hours are used to calculate the radiation doses at the site boundary for a 2-hour exposure, and the releases for 0 to 8 hours are used to calculate the radiation doses at the low population zone and to the operators in the control room for the duration of the accident.

In the LOFTTR2 analyses, the SGTR recovery actions in the E-3 guideline were simulated until the termination of primary-to-secondary leakage. After the primary-to-secondary leakage is terminated, the operators will continue the SGTR recovery actions to prepare the plant for cooldown to cold shutdown conditions. When these recovery actions are completed, the plant should be cooled and depressurized to cold shutdown conditions. In accordance with the methodology in Reference 2 it was assumed that the cooldown is performed using HNP Emergency Operating Procedures (EOPs) ES-3.3, Post-SGTR Cooldown Using Steam Dump, since this method results in a conservative evaluation of the long-term releases for the offsite dose analysis compared to the other cooldown methods in the EOPs. This procedure for depressurizing the ruptured steam generator is assumed even though the LOFTTR2 analysis performed to calculate releases up until break flow termination has assumed PORV isolation.

The high level actions for the post-SGTR cooldown method using steam dump in the HNP EOP ES-3.3 are discussed below.

1. Prepare for Cooldown to Cold Shutdown.

   The initial steps to prepare for cooldown to cold shutdown will be continued if they have not already been completed. A few additional steps are also performed prior to initiating cooldown. These include isolating the cold leg SI accumulators to prevent unnecessary injection, energizing pressurizer heaters as necessary to saturate the pressurizer water and to provide for better pressure control, and assuring shutdown margin in the event of a potential boron dilution due to in-leakage from the ruptured steam generator.

2. Cooldown RCS to Residual Heat Removal (RHR) System Temperature.

   The RCS is cooled by steaming and feeding the intact steam generators similar to a normal cooldown. Since all immediate safety concerns have been resolved, the cooldown rate should be maintained less than the maximum allowable rate of 100°F/hr. The preferred means for cooling the RCS is steam dump to the condenser, since this minimizes the radiological releases and conserves feedwater supply. The PORVs on the intact steam generators can also be used if steam dump to the condenser is unavailable. Since a loss-of-offsite power is assumed, an assumption was made that the cooldown is performed using steam dump to the atmosphere via the intact steam generators' PORVs. When the RHR system operating temperature is reached, the cooldown is stopped until RCS pressure can also be decreased. This ensures that pressure/temperature limits will not be exceeded.

3. Depressurize RCS to RHR System Pressure.

When the cooldown to RHR system temperature is completed, the pressure in the ruptured steam generator is decreased by releasing steam from the ruptured steam generator. Steam release to the condenser is preferred, since this minimizes radiological releases, but steam can be released to the atmosphere using the PORV on the ruptured steam generator if the condenser is not available. Consistent with the assumption of a loss-of-offsite power, it was assumed that the ruptured steam generator is depressurized by releasing steam via the PORV. As the ruptured steam generator pressure is reduced, the RCS pressure is maintained equal to the pressure in the ruptured steam generator in order to prevent in-leakage of secondary side water or additional primary to secondary leakage. Although normal pressurizer spray is the preferred means of RCS pressure control, auxiliary spray or pressurizer PORV can be used to control RCS pressure if pressurizer spray is not available.

4. Cooldown to Cold Shutdown.

When RCS temperature and pressure have been reduced to the RHR system in-service values, RHR system cooling is initiated to complete the cooldown to cold shutdown. When cold shutdown conditions are achieved, the pressurizer can be cooled to terminate the event.

6.3.2.3 Description of Analysis Cases

The LOFTTR2 analysis results for the HNP offsite dose evaluation are described below, considering operation at the uprated NSSS power of 2912.4 MWt with Model Δ75 replacement steam generators. The sequence of events for the analysis is presented in Table 6.3.2-1. The transient results for this case are similar to the transient results for the overfill analysis until the ruptured steam generator is isolated. The transient behavior is different after this time, as it is assumed that the ruptured steam generator PORV fails open at that time.

Following the tube rupture, the RCS pressure decreases as shown in Figure 6.3.2-1 due to the primary to secondary leakage. In response to this depressurization, the reactor trips on overtemperature-ΔT at about 114 seconds. After reactor trip, core power rapidly decreases to decay heat levels and the RCS depressurization becomes more rapid. The steam dump system is inoperable due to the assumed loss-of-offsite power, which results in the secondary pressure rising to the steam generator PORV setpoint as shown in Figure 6.3.2-2. The RCS pressure and pressurizer level also decrease more rapidly following reactor trip as shown in Figures 6.3.2-1 and 6.3.2-3. The decreasing pressurizer pressure leads to an automatic SI signal on low pressurizer pressure at approximately 178 seconds.

Major Operator Actions

1. Identify and Isolate the Ruptured Steam Generator.

   Recovery actions begin by throttling the auxiliary feedwater flow to the ruptured steam generator and isolating steam flow from the ruptured steam generator. As indicated previously, isolation of the AFW flow to the ruptured steam generator was assumed to be completed at 10 minutes after the initiation of the SGTR or when the narrow range level reaches 30 percent, whichever time is greater. For the HNP analysis, the time to reach 30 percent is less than 10 minutes, and thus the ruptured steam generator is assumed to be isolated at 10 minutes. Also, as indicated previously, complete isolation of steam flow from the ruptured steam generator is verified when the narrow range level reaches 30 percent on the ruptured steam generator or at 12 minutes after initiation of the SGTR, whichever is longer. For the HNP analysis, the time to reach 30 percent is less than 12 minutes, and thus the ruptured steam generator is assumed to be isolated at 12 minutes.

   The ruptured steam generator PORV is also assumed to fail open at this time. The failure causes the steam generator to rapidly depressurize, which results in an increase in primary to secondary leakage. The depressurization of the ruptured steam generator increases the break flow and energy transfer from primary to secondary, which results in RCS pressure and temperature decreasing more rapidly than in the margin to overfill analysis. The ruptured steam generator depressurization causes a cooldown in the intact steam generators loops. It is assumed that the time required for the operator to identify that the ruptured steam generator PORV is open and to locally close the associated block valve is 20 minutes. At 1922 seconds the depressurization of the ruptured steam generator is terminated and the ruptured steam generator pressure begins to increase as shown in Figure 6.3.2-2.

2. Cooldown the RCS to establish Subcooling Margin.

   After the block valve for the ruptured steam generator PORV is closed, there is a 5 minute operator action time imposed prior to initiation of cooldown. The depressurization of the ruptured steam generator due to the failed-open PORV affects the RCS cooldown target temperature since the temperature is determined based upon the pressure in the ruptured steam generator at that time. Since offsite power is lost, the RCS is cooled by dumping steam to the atmosphere using the intact steam generators' PORVs. The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance of 20°F for instrument uncertainty. Because of the lower pressure in the ruptured steam generator when the cooldown is initiated (relative to the margin to overfill analysis), the associated temperature to which the RCS must be cooled is also lower. However, both intact steam generator PORVs are available for the cooldown in this case, whereas in the margin to overfill analysis the limiting single failure resulted in only one intact steam

generator being available for the cooldown. The cooldown begins at 2224 seconds and is completed at 2996 seconds.

The reduction in the intact steam generators' pressure required to accomplish the cooldown is shown in Figure 6.3.2-2 and the effect of the cooldown on the RCS temperatures is shown in Figures 6.3.2-4 and 6.3.2-5. The RCS pressure and pressurizer level also decrease during this cooldown process due to shrinkage of the reactor coolant as shown in Figures 6.3.2-1 and 6.3.2-3. The break flow flashing fraction is calculated throughout the transient based on the difference between the enthalpy of the break flow and the saturation enthalpy at the ruptured steam generator pressure as shown in Figure 6.3.2-7. Break flow is calculated to stop flashing at approximately 2500 seconds as a result of the reduction in primary coolant temperature associated with the cooldown (Figure 6.3.2-4) and the increase in ruptured steam generator pressure following isolation of the failed open PORV (Figure 6.3.2-2).

3. Depressurize to Restore Inventory.

After the RCS cooldown is completed, a 240 second operator action time is included prior to the RCS depressurization. The RCS depressurization is performed to assure adequate coolant inventory prior to terminating SI flow. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS is depressurized by opening a pressurizer PORV. The RCS depressurization is initiated at 3236 seconds and continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than the allowance of 10 percent for pressurizer level uncertainty, or pressurizer level is greater than 75 percent, or RCS subcooling is less than the 20°F allowance for subcooling uncertainty. For this case, the RCS depressurization is terminated at 3312 seconds because the RCS pressure is reduced to less than the ruptured steam generator pressure and the pressurizer level is above 10 percent. The RCS depressurization reduces the break flow as shown in Figure 6.3.2-6 and increases SI flow to refill the pressurizer, as shown in Figure 6.3.2-3.

4. Terminate SI to Stop Primary to Secondary Leakage.

The previous actions establish adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent re-pressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated at this time if RCS subcooling is greater than the 20°F allowance for subcooling uncertainty, minimum AFW flow is available or at least one intact steam generator level is in the narrow range, the RCS pressure is stable or increasing, and the pressurizer level is greater than the 10 percent allowance for uncertainty.

After depressurization is completed, an operator action time of 3 minutes was assumed prior to SI termination. Since the above requirements are satisfied, SI termination actions were performed at 3492 seconds by closing off the SI flow path. After SI termination the RCS pressure begins to decrease as shown in Figure 6.3.2-1.

The intact steam generators' PORVs also automatically open to dump steam to maintain the prescribed RCS temperature to ensure that subcooling is maintained. When the PORVs are opened, the increased energy transfer from primary to secondary also aids in the depressurization of the RCS to the ruptured steam generator pressure. The ruptured steam generator pressure increases to the PORV setpoint and steam release is reinitiated. Steam generator pressure is maintained at the steam generator PORV setpoint rather than the safety valve setpoint for modeling efficiency. This modeling is conservative since it delays break flow termination by requiring the RCS pressure to drop further, maximizes the break flow rate by maintaining a larger primary-to-secondary pressure differential, and results in more steam release from the ruptured steam generator. The primary to secondary leakage continues after the SI flow is terminated until the RCS and ruptured steam generator pressures equalize.

Calculation of Mass Releases

The operator actions for the SGTR recovery up to the termination of primary to secondary leakage are simulated in the LOFTTR2 analyses. Thus, the steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and the primary to secondary leakage into the ruptured steam generator were determined from the LOFTTR2 results for the period from the initiation of the accident until the leakage is terminated.

Following the termination of leakage, it was assumed that the RCS and intact steam generators conditions are maintained stable for a 20 minute period until the cooldown to cold shutdown is initiated. The PORVs for the intact steam generators were then assumed to be used to cool down the RCS to the RHR system operating temperature of 325°F, at the maximum allowable cooldown rate of 100°F/hr. The RCS and the intact steam generators temperatures at 2 hours were then determined using the RCS and intact steam generators parameters at the time of leakage termination and the RCS cooldown rate. The steam releases and the feedwater flows for the intact steam generators for the period from leakage termination until 2 hours were determined from a mass and energy balance using the calculated RCS and intact steam generators conditions at the time of leakage termination and at 2 hours. Since the ruptured steam generator is isolated, no change in the ruptured steam generator conditions is assumed to occur until subsequent depressurization.

The RCS cooldown was assumed to be continued after 2 hours until the RHR system in-service temperature of 325°F is reached. Depressurization of the ruptured steam generator was then assumed to be performed immediately following the completion of the RCS cooldown. The ruptured steam generator was assumed to be depressurized to the RHR in-service pressure of

365 psia via steam release from the ruptured steam generator PORV, since this maximizes the steam release from ruptured steam generator to the atmosphere which is conservative for the evaluation of the offsite radiation doses. The RCS pressure is also assumed to be reduced concurrently as the ruptured steam generator is depressurized. It is assumed that the continuation of the RCS cooldown and depressurization to RHR operating conditions are completed within 8 hours after the accident since there is ample time to complete the operations during this time period. The steam releases and feedwater flows from 2 to 8 hours were determined for the intact steam generators from a mass and energy balance using the RCS and steam generator conditions at 2 hours and at the RHR system in-service conditions. The steam released from the ruptured steam generator from 2 to 8 hours was determined based on a mass and energy balance for the ruptured steam generator using the conditions at the time of leakage termination and saturated conditions at the RHR in-service pressure.

After 8 hours, it is assumed that further plant cooldown to cold shutdown as well as long-term cooling is provided by the RHR system. Therefore, the steam releases to the atmosphere are terminated after RHR in-service conditions are assumed to be reached at 8 hours.

The analysis was also run at the current NSSS power level of 2787.4 MWt, since the assumed power level impacts the calculated offsite and control room doses.

6.3.2.4 Acceptance Criteria

The analysis is performed to calculate the mass transfer data for input to the radiological consequences analysis. As such no acceptance criteria are defined. The results of the analysis are used as input to the radiological consequences analysis presented in Section 6.3.3.

6.3.2.5 Results

LOFTTR2 Analysis Results

The primary to secondary break flow rate throughout the recovery operations is presented in Figure 6.3.2-6. The calculated break flow flashing fraction and integrated flashed break flow are presented in Figures 6.3.2-7 and 6.3.2-8, respectively. The ruptured steam generator PORV steam release rate is presented in Figure 6.3.2-9. The total intact steam generator PORV steam release rate is presented in Figure 6.3.2-10. The ruptured steam water volume is shown in Figure 6.3.2-11. For this case, the water volume in the ruptured steam generator when the break flow is terminated is less than the volume for the margin to overfill case and significantly less than the total steam generator volume of 5545 ft^3. The ruptured steam water mass is shown in Figure 6.3.2-12.

The results for the analysis performed with the replacement steam generators at the current NSSS power level of 2787.4 MWt progress in the same manner. The sequence of events for the analysis is included in Table 6.3.2-1. The primary to secondary break flow rate throughout the

recovery operations is presented in Figure 6.3.2-13. The calculated break flow flashing fraction and integrated flashed break flow are presented in Figures 6.3.2-14 and 6.3.2-15, respectively. The ruptured steam generator PORV steam release rate is presented in Figure 6.3.2-16. The total intact steam generator PORV steam release rate is presented in Figure 6.3.2-17.

Mass Release Results

The mass release calculations were performed using the methodology discussed above. For the time period from initiation of the accident until leakage termination, the releases were determined from the LOFTTR2 results for the time prior to reactor trip and following reactor trip. Since the condenser is in service until reactor trip, any radioactivity released to the atmosphere prior to reactor trip will be through the condenser vacuum exhaust. After reactor trip, the releases to the atmosphere are assumed to be via the steam generator PORVs.

The mass releases for the SGTR event assuming failure and isolation of the ruptured steam generator PORV are presented in Table 6.3.2-2, for the analysis modeling the Model Δ75 replacement steam generators and the uprated NSSS power of 2912.4 MWt. The results indicate that approximately 138,300 lbm of steam is released to the atmosphere from the ruptured steam generator within the first 2 hours. After 2 hours, 35,100 lbm of steam is released to the atmosphere from the ruptured steam generator. A total of 167,900 lbm of primary water is transferred to the secondary side of the ruptured steam generator before break flow is terminated. A total of 10,843 lbm of this break flow is assumed to flash to steam upon entering the steam generator.

The mass releases for the analysis modeling the Model Δ75 replacement steam generators and the current NSSS power of 2787.4 MWt are presented in Table 6.3.2-3. The results indicate that approximately 135,700 lbm of steam is released to the atmosphere from the ruptured steam generator within the first 2 hours. After 2 hours, 34,400 lbm of steam is released to the atmosphere from the ruptured steam generator. A total of 165,400 lbm of primary water is transferred to the secondary side of the ruptured steam generator before break flow is terminated. A total of 10,598 lbm of this break flow is assumed to flash to steam upon entering the steam generator.

6.3.2.6 Conclusions

The analysis performed to calculate the mass transfer data for input to the radiological consequences analysis has been completed and data tabulated for the limiting case. The results of the analysis are used as input to the radiological consequences analysis presented in Section 6.3.3.

6.3.2.7 References

1. WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," Lewis, Huang, Behnke, Fittante, Gelman, August 1987.

2. Supplement 1 to WCAP-10698-P-A, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," Lewis, Huang, Rubin, March 1986.

3. WCAP-12403, "LOFTR2 Analysis for a Steam Generator Tube Rupture with Revised Operator Action Times for Shearon Harris Nuclear Power Plant," Huang, Lewis, Marmo, Rubin, November 1989.

4. Supplement 1 to WCAP-12403, "Steam Generator Tube Rupture Analysis for Shearon Harris Nuclear Power Plant," Lewis, Lowe, Monahan, Rubin, Tanz, November 1992.

Table 6.3.2-1 Sequence of Events for Offsite Radiation Dose Analysis		
	Operation at 2912.4 MWt	**Operation at 2787.4 MWt**
Event	**Time (seconds)**	**Time (seconds)**
Steam Generator Tube Rupture	0	0
Reactor Trip	114	114
SI Actuation	178	170
AFW Flow to Ruptured SG Isolated	600	600
Ruptured SG Isolated	720	720
Ruptured SG PORV Fails Open	722	722
Ruptured SG Block Valve Closed	1922	1922
RCS Cooldown Initiated	2224	2224
RCS Cooldown Terminated	2996	2968
RCS Depressurization Initiated	3236	3208
RCS Depressurization Terminated	3312	3288
SI Terminated	3492	3468
Break Flow Terminated	4652	4468

Table 6.3.2-2 Mass Releases for Operation At 2912.4 MWt; Total Mass Flow (Pounds)				
	Time Period			
	Time Zero to Time of Reactor Trip*	Time of Reactor Trip to Time at Which Break Flow is Terminated*	Time at Which Break Flow is Terminated to 2 Hours	2 Hours to Time at Which RCS Reaches RHR In-Service Conditions*
Ruptured SG				
- Condenser	128,300	0	0	0
- Atmosphere	0	138,300	0	35,100
- Feedwater	123,400	33,000	0	0
Intact SGs				
- Condenser	254,100	0	0	0
- Atmosphere	0	176,900	183,300	862,800
- Feedwater	254,100	292,400	201,800	894,900
Break Flow	4900	163,000	0	0
Flashed Break Flow	830	10,013	0	0

* Reactor trip occurs at 114 seconds; break flow is terminated at 4652 seconds; RHR conditions are reached at 8 hours.

Table 6.3.2-3 Mass Releases for Operation At 2787.4 MWt; Total Mass Flow(Pounds)				
	Time Period			
	Time Zero to Time of Reactor Trip*	Time of Reactor Trip to Time at Which Break Flow is Terminated*	Time at Which Break Flow is Terminated to 2 Hours	2 Hours to Time at Which RCS Reaches RHR In-Service Conditions*
Ruptured SG				
- Condenser	122,900	0	0	0
- Atmosphere	0	135,700	0	34,400
- Feedwater	118,000	32,500	0	0
Intact SGs				
- Condenser	243,400	0	0	0
- Atmosphere	0	166,500	194,100	823,100
- Feedwater	243,400	281,000	215,000	850,600
Break Flow	4,800	160,600	0	0
Flashed Break Flow	794	9,804	0	0

* Reactor trip occurs at 114 seconds; break flow is terminated at 4468 seconds; RHR conditions are reached at 8 hours.

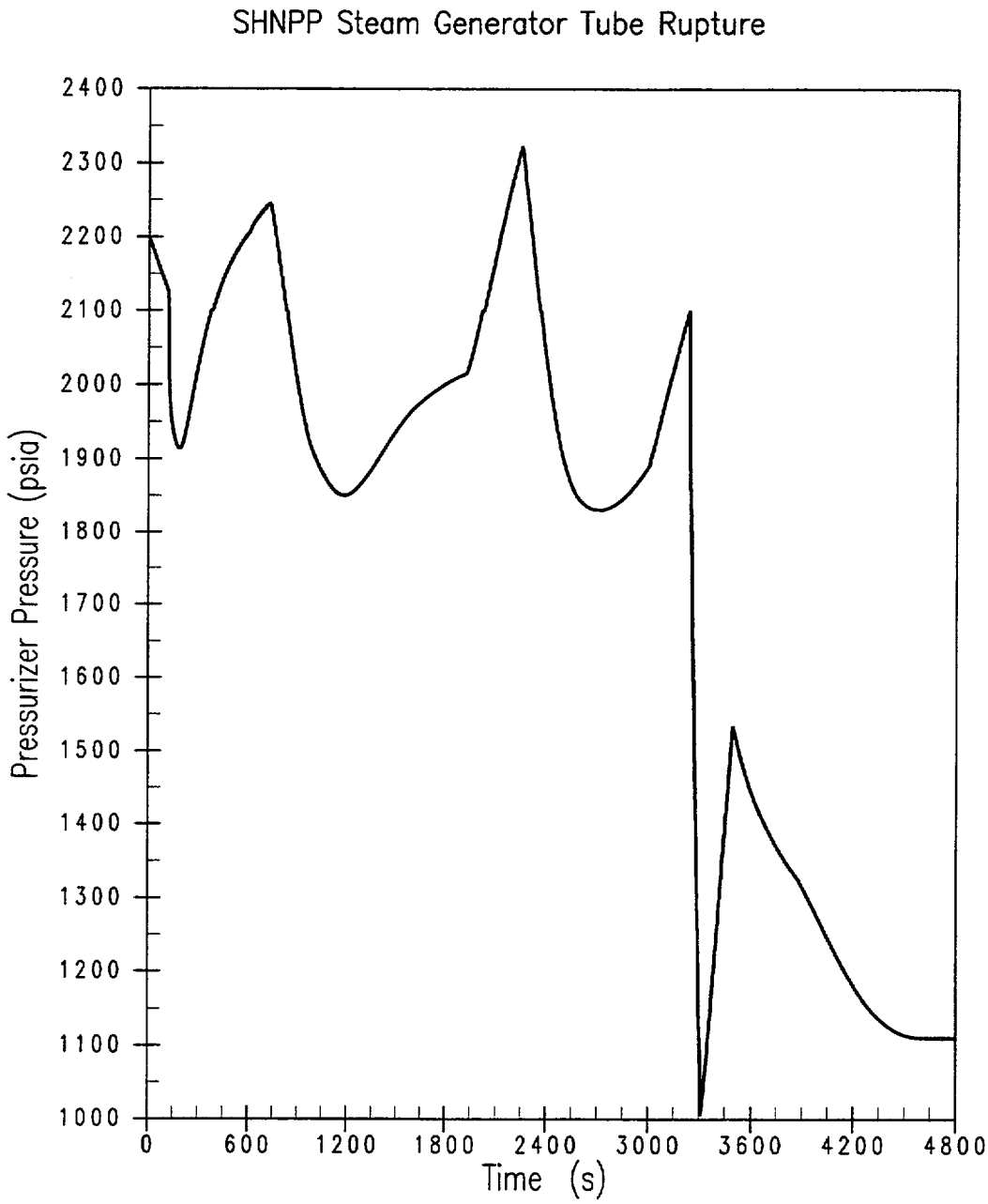

Figure 6.3.2-1
Pressurizer Pressure – Offsite Radiation Dose Analysis (2912.4 MWt)

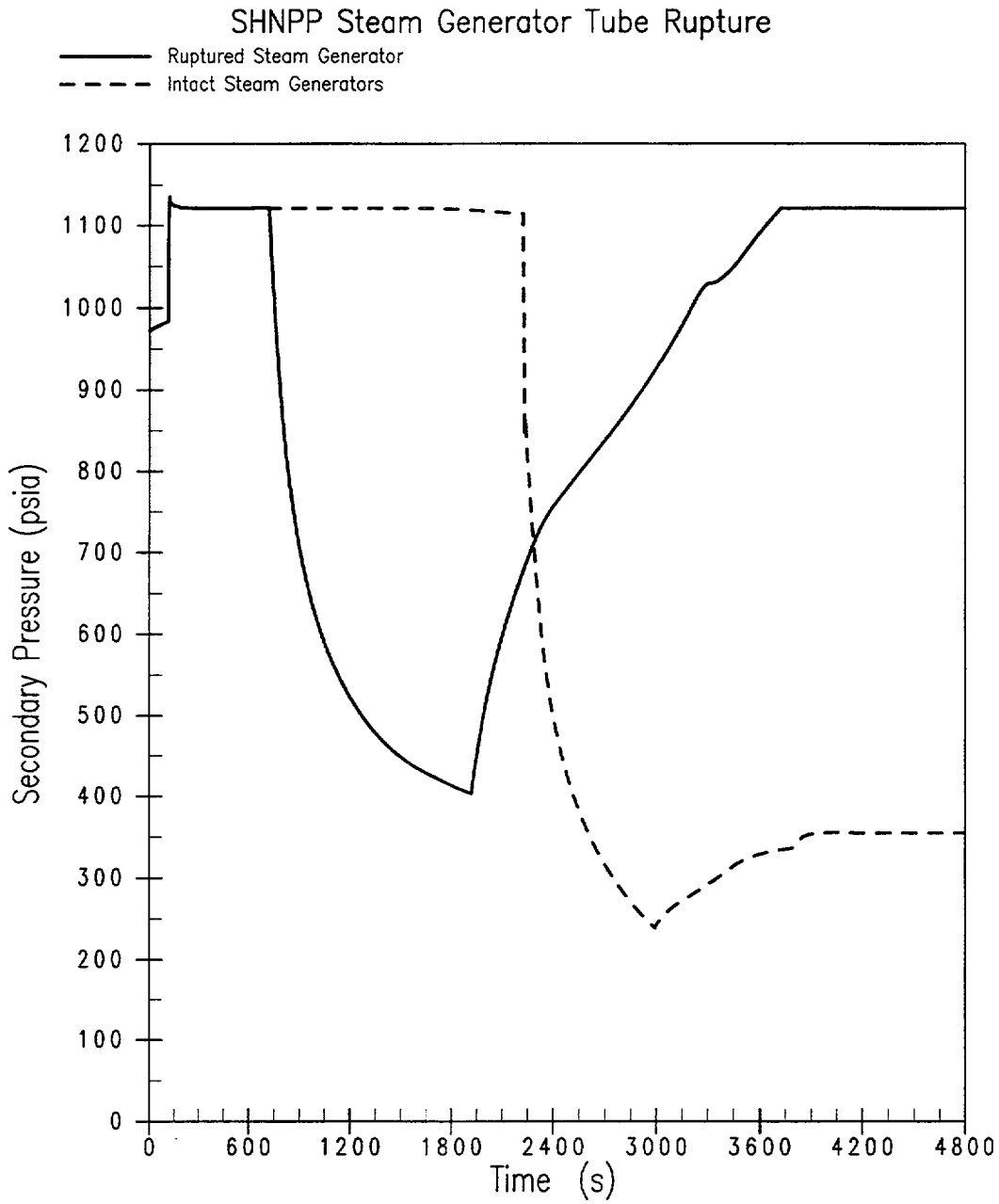

Figure 6.3.2-2
Secondary Pressure – Offsite Radiation Dose Analysis (2912.4 MWt)

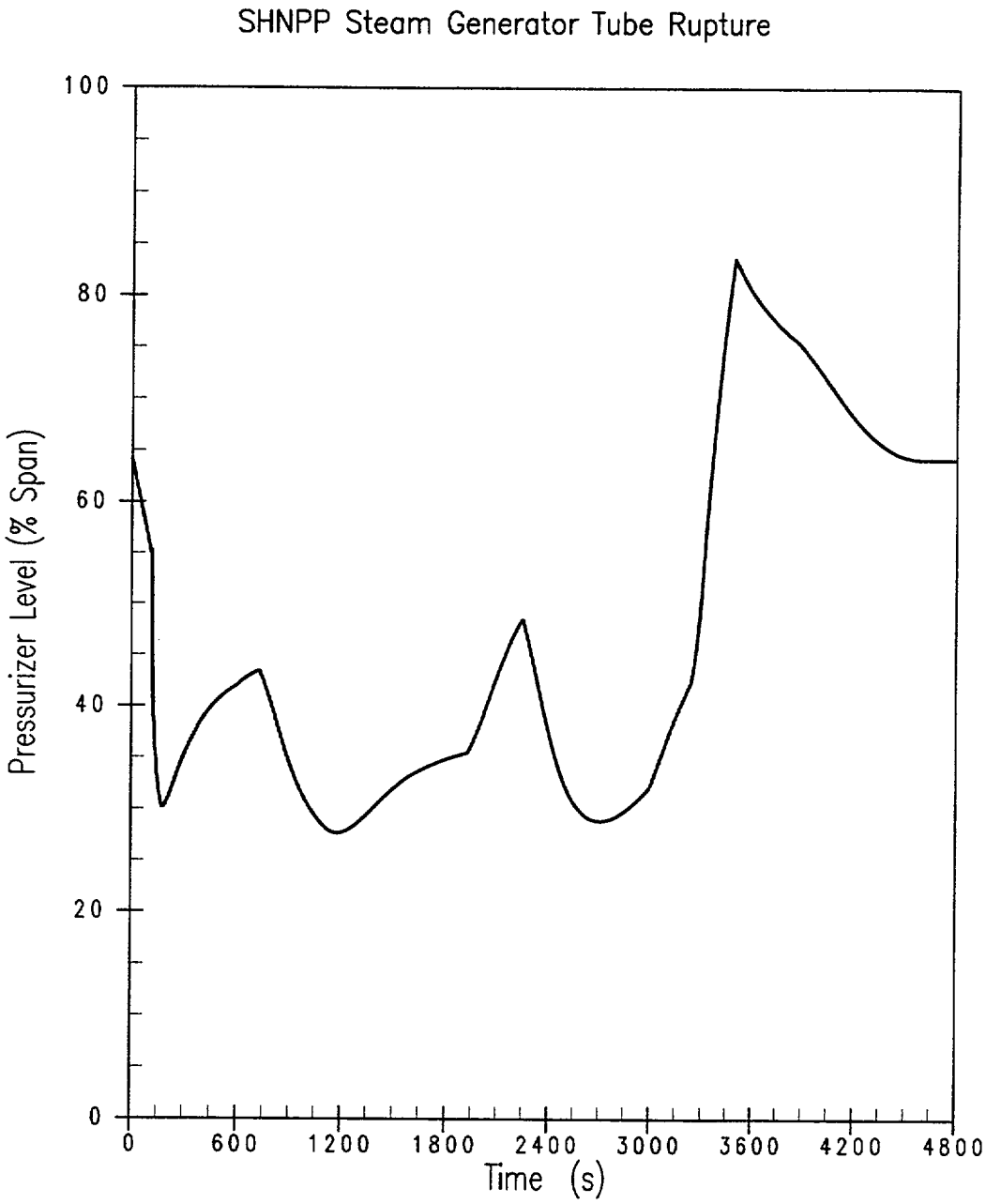

Figure 6.3.2-3
Pressurizer Level – Offsite Radiation Dose Analysis (2912.4 MWt)

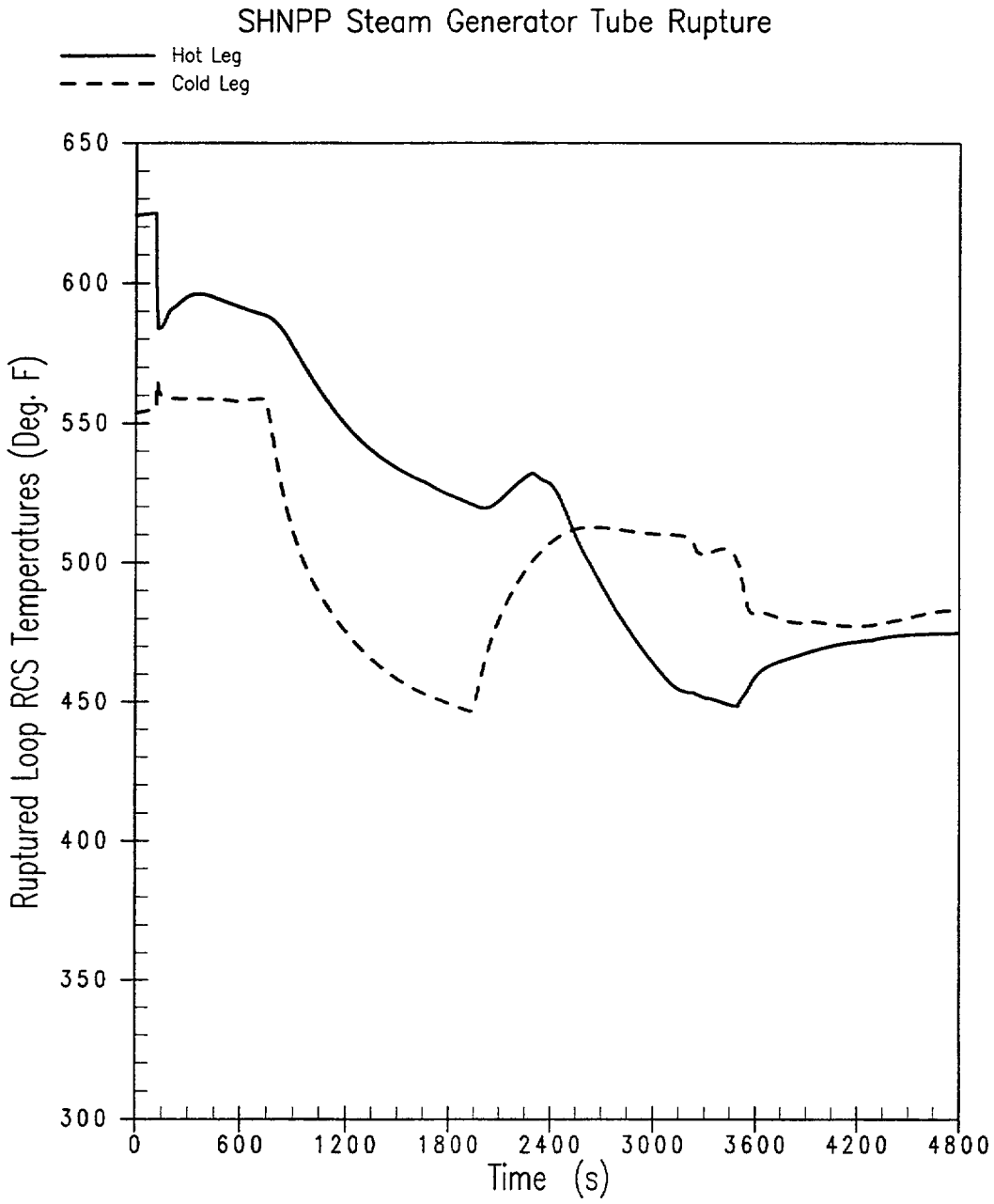

Figure 6.3.2-4
Ruptured Loop Hot & Cold Leg Temperatures –
Offsite Radiation Dose Analysis (2912.4 MWt)

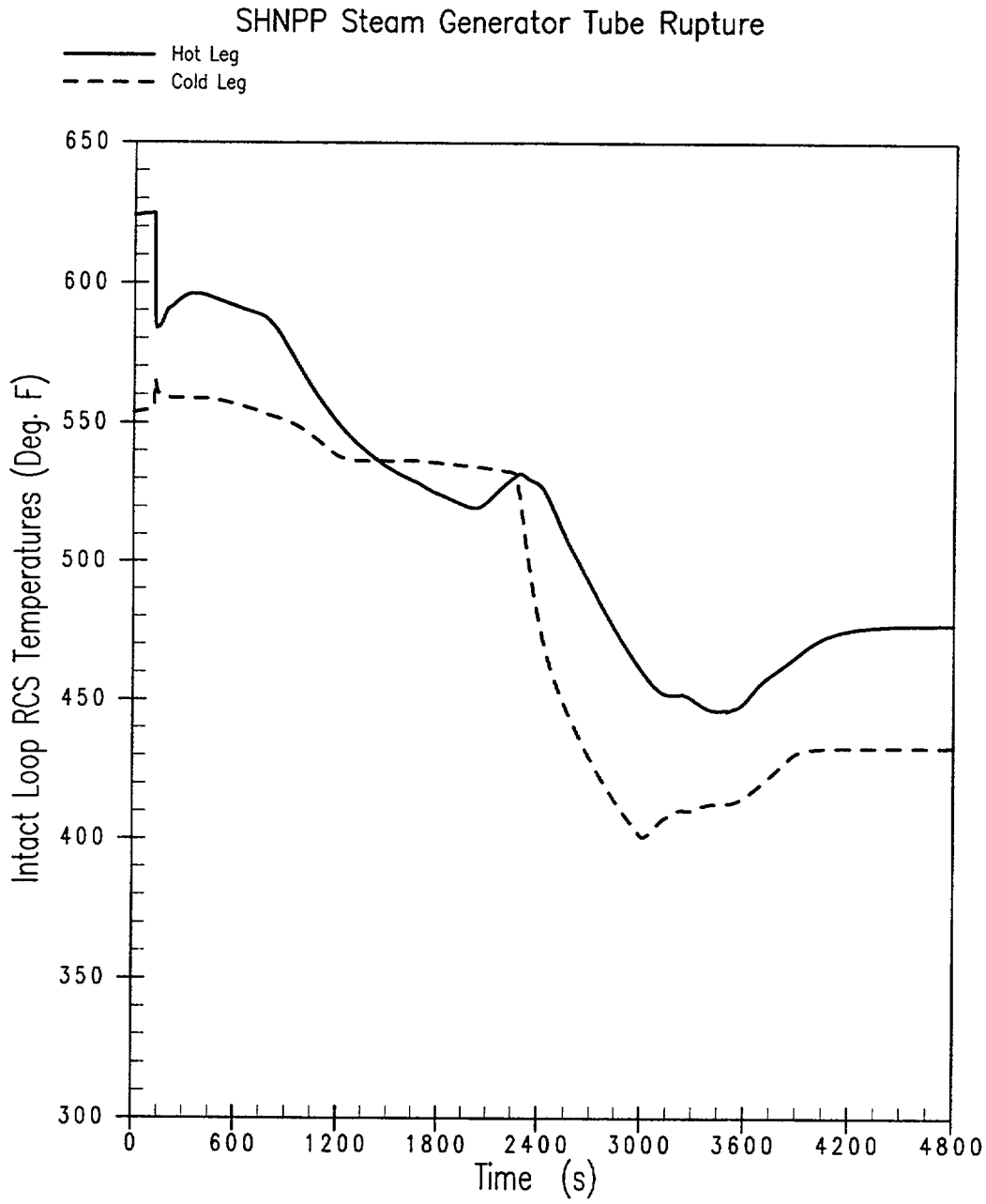

Figure 6.3.2-5
Intact Loop Hot & Cold Leg Temperatures –
Offsite Radiation Dose Analysis (2912.4 MWt)

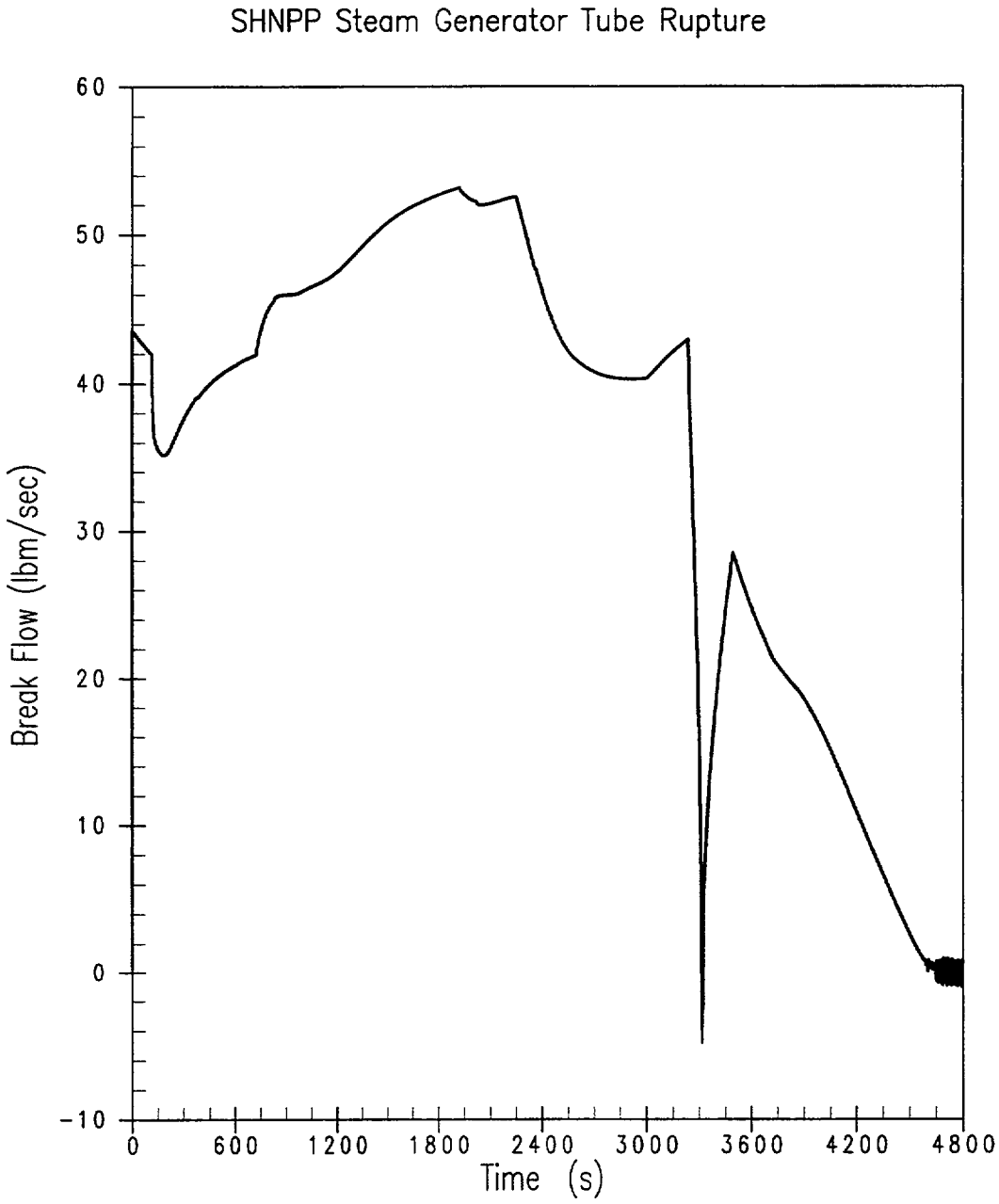

Figure 6.3.2-6
Primary to Secondary Break Flow –
Offsite Radiation Dose Analysis (2912.4 MWt)

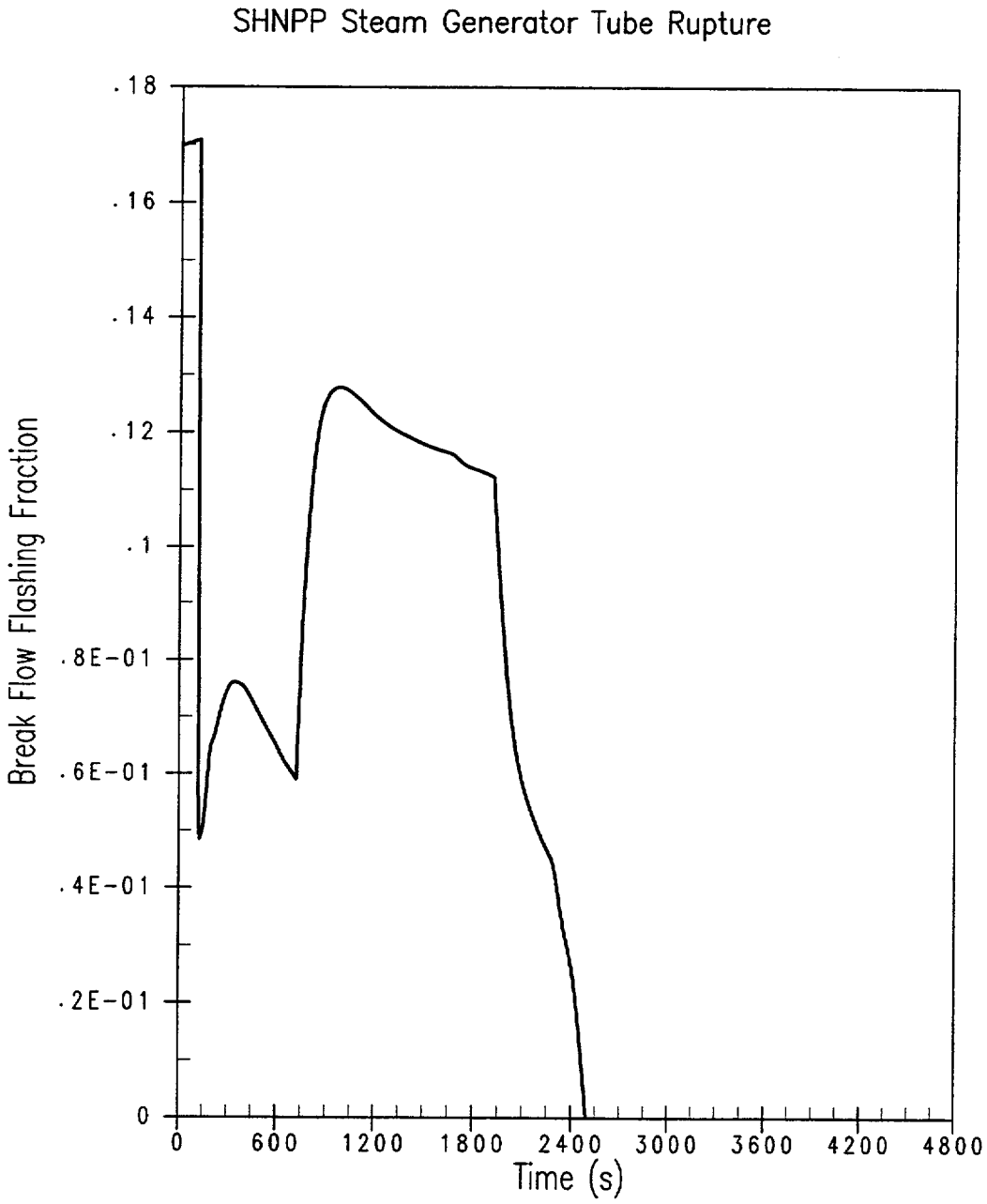

Figure 6.3.2-7
Break Flow Flashing Fraction –
Offsite Radiation Dose Analysis (2912.4 MWt)

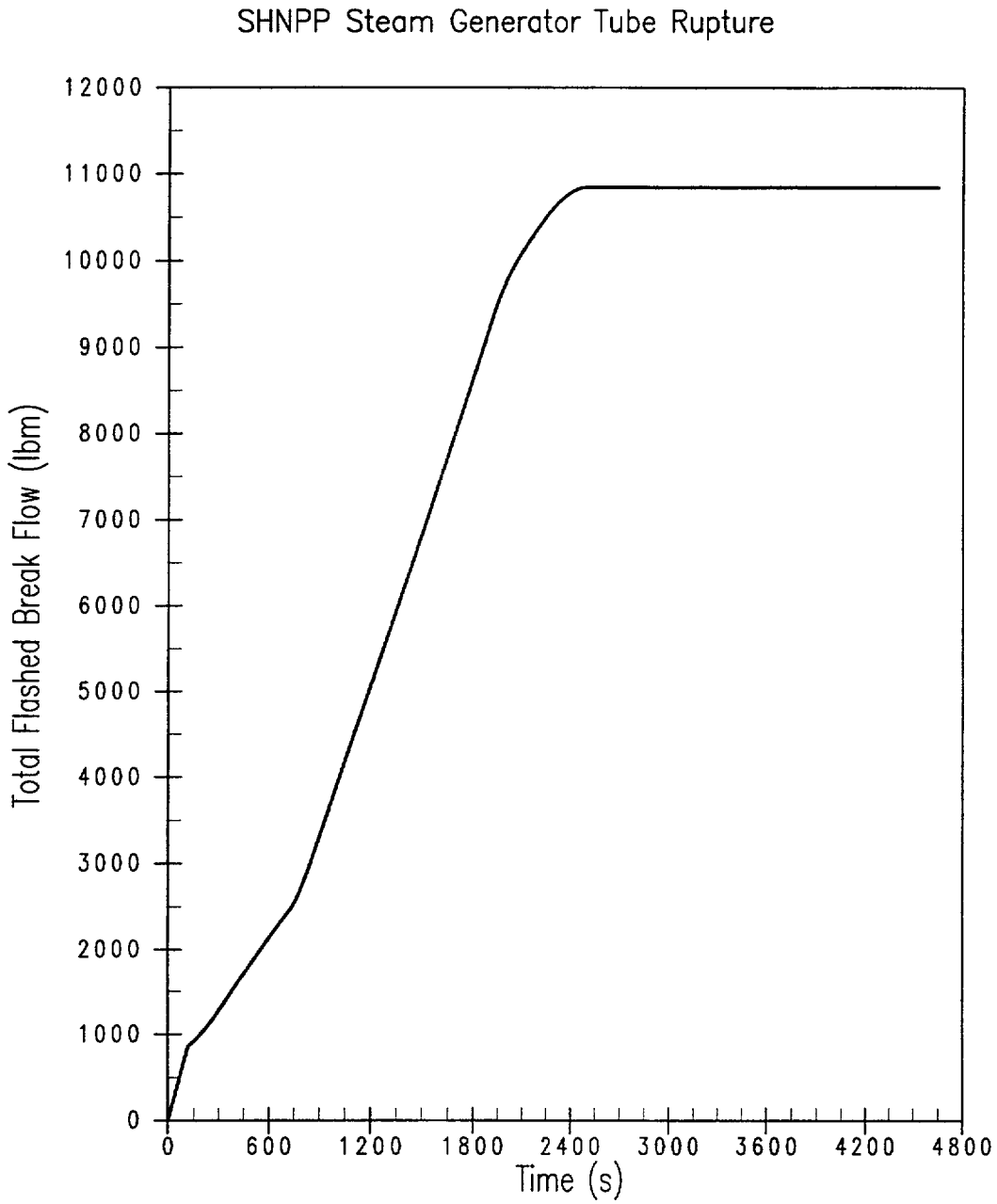

Figure 6.3.2-8
Total Flashed Break Flow –
Offsite Radiation Dose Analysis (2912.4 MWt)

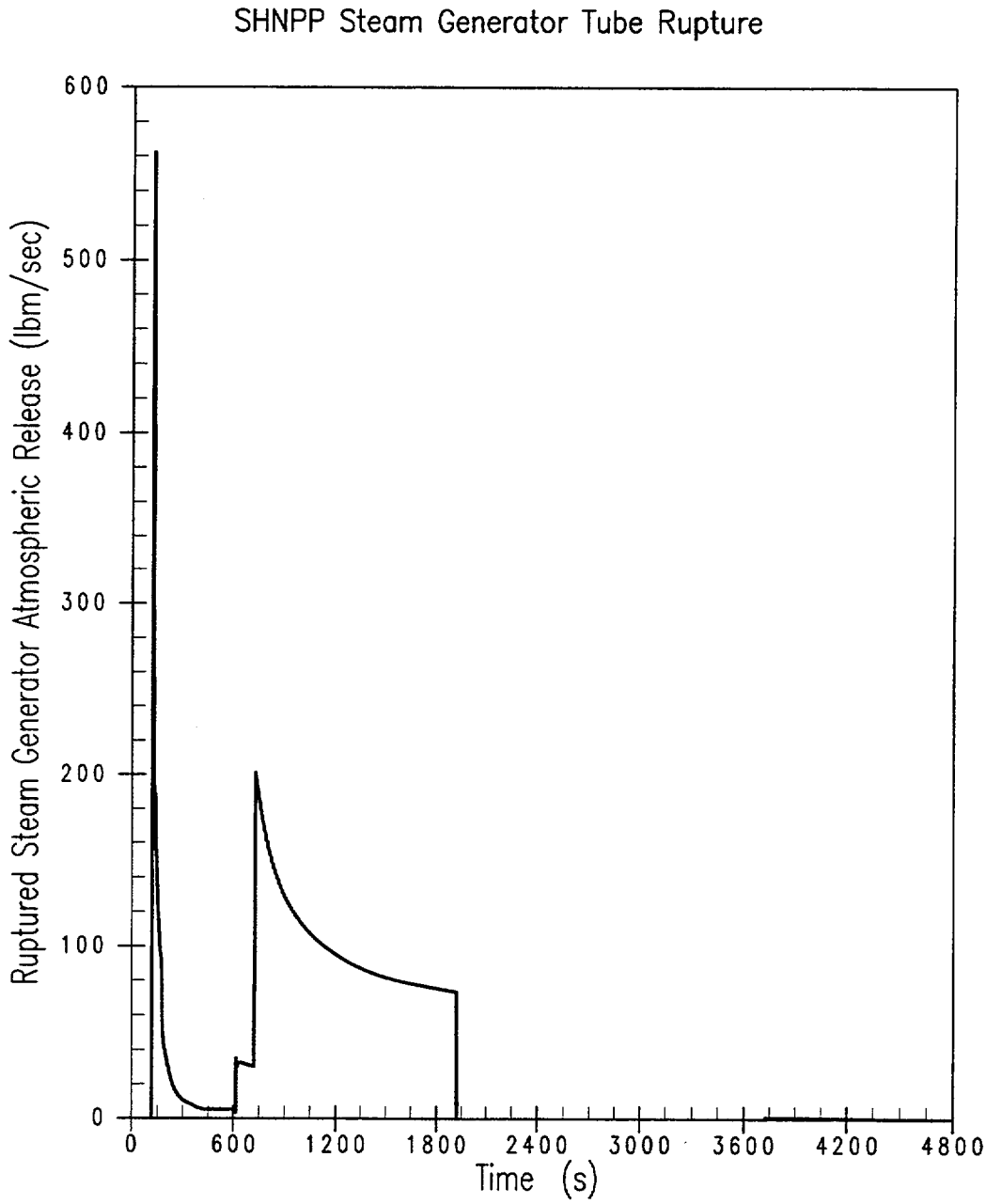

Figure 6.3.2-9
Ruptured SG Mass Release Rate to the Atmosphere – Offsite Radiation Dose Analysis (2912.4 MWt)

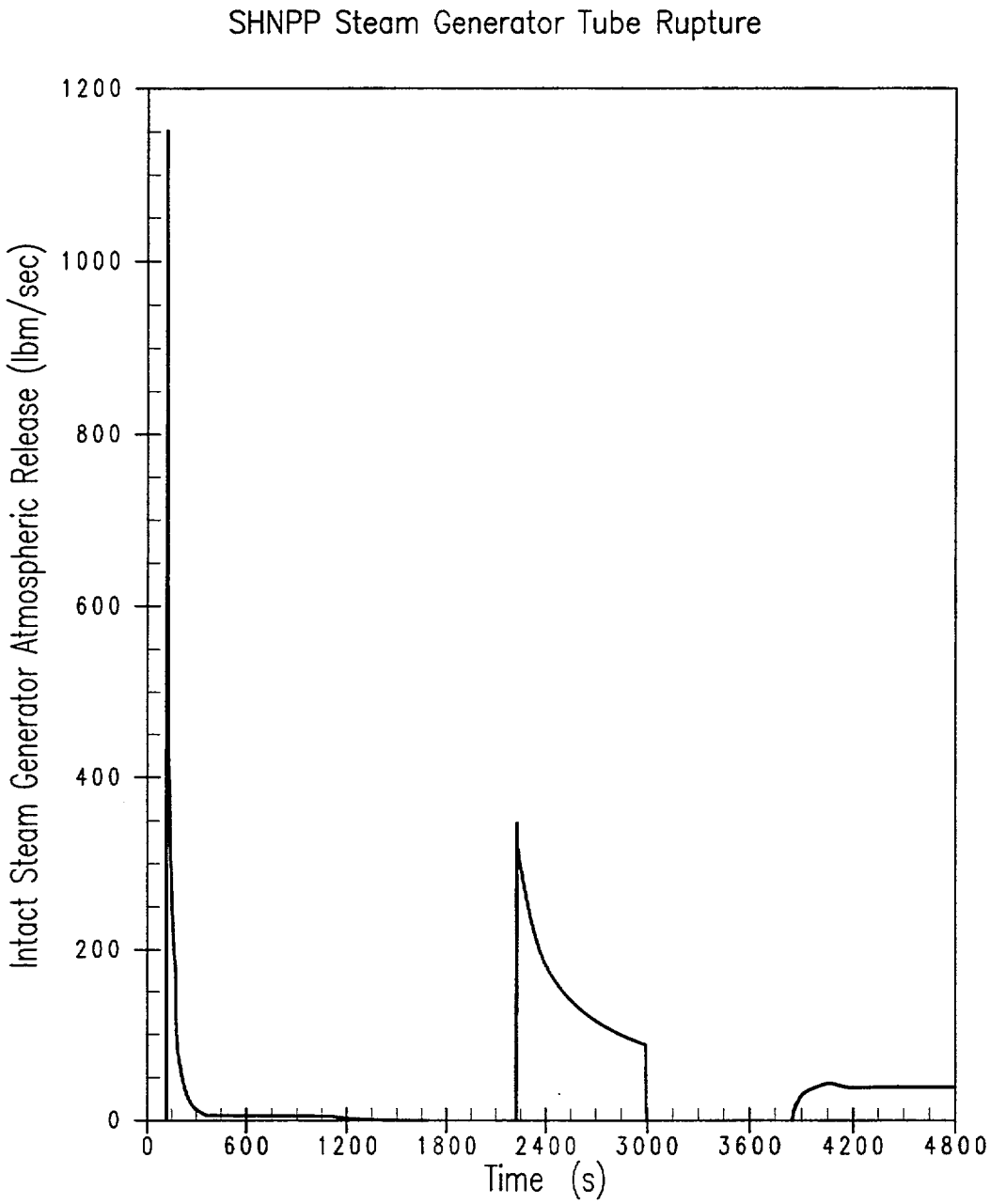

Figure 6.3.2-10
Intact SGs Mass Release Rate to the Atmosphere –
Offsite Radiation Dose Analysis (2912.4 MWt)

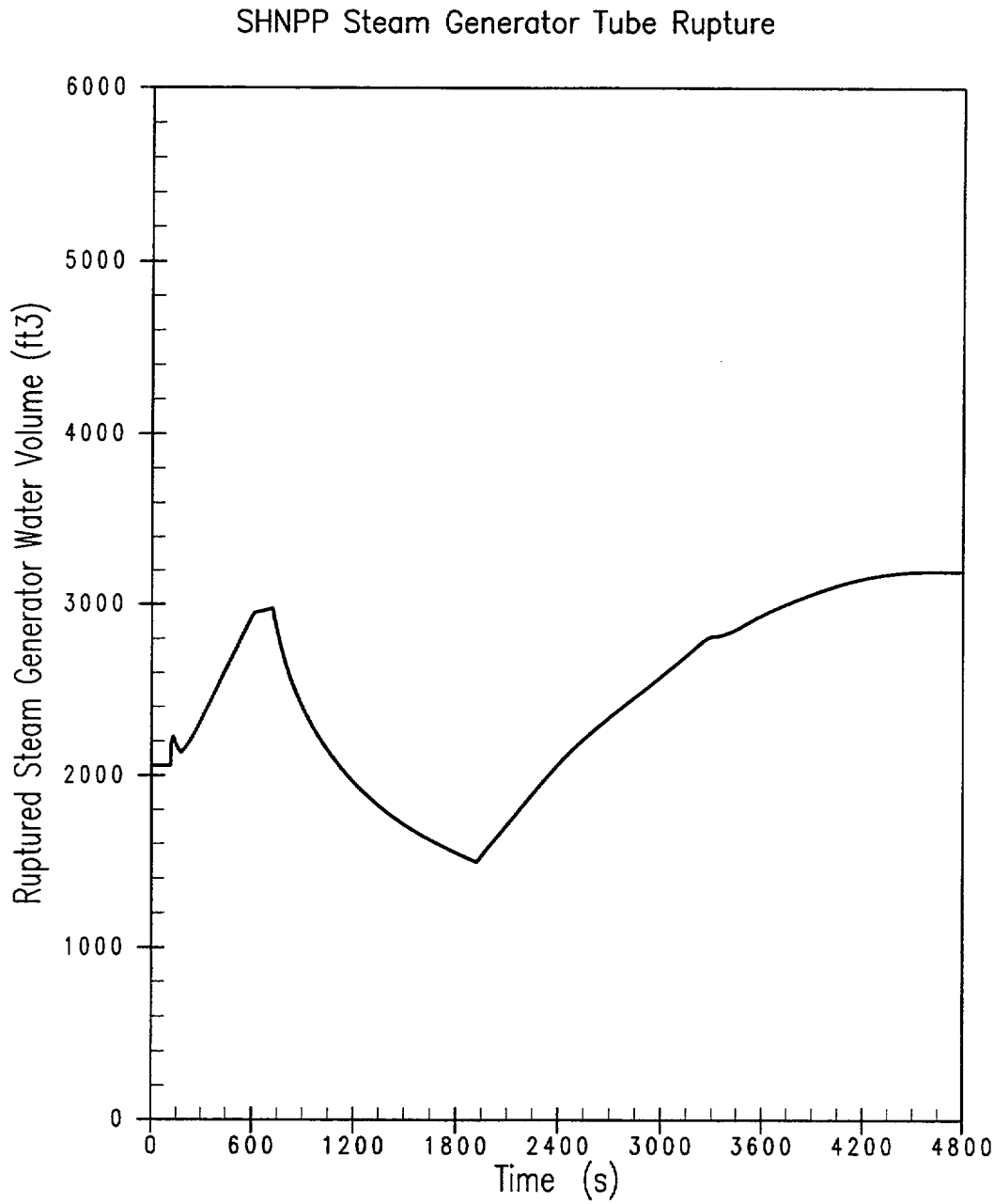

Figure 6.3.2-11
Ruptured SG Water Volume –
Offsite Radiation Dose Analysis (2912.4 MWt)

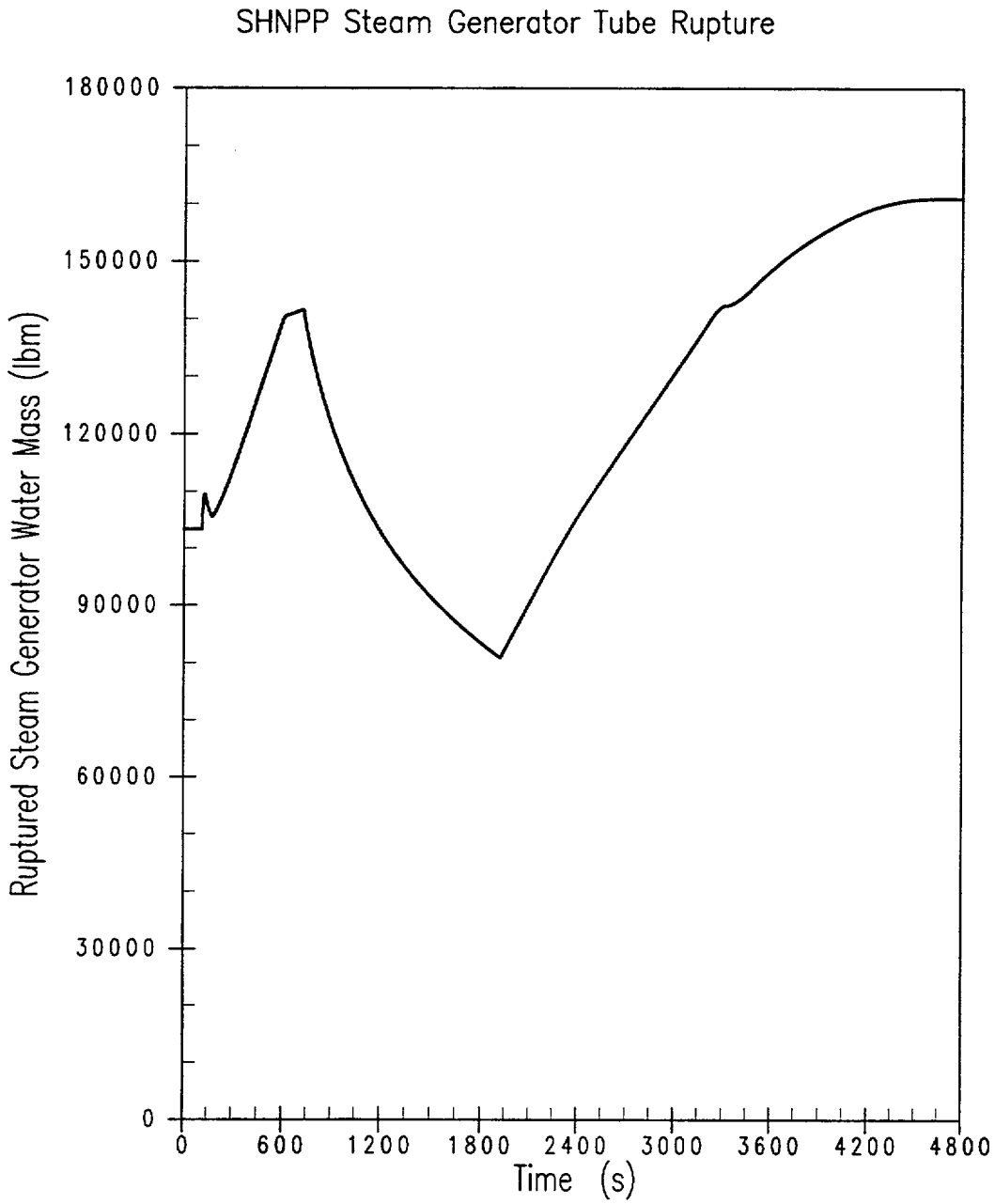

Figure 6.3.2-12
Ruptured SG Water Mass –
Offsite Radiation Dose Analysis (2912.4 MWt)

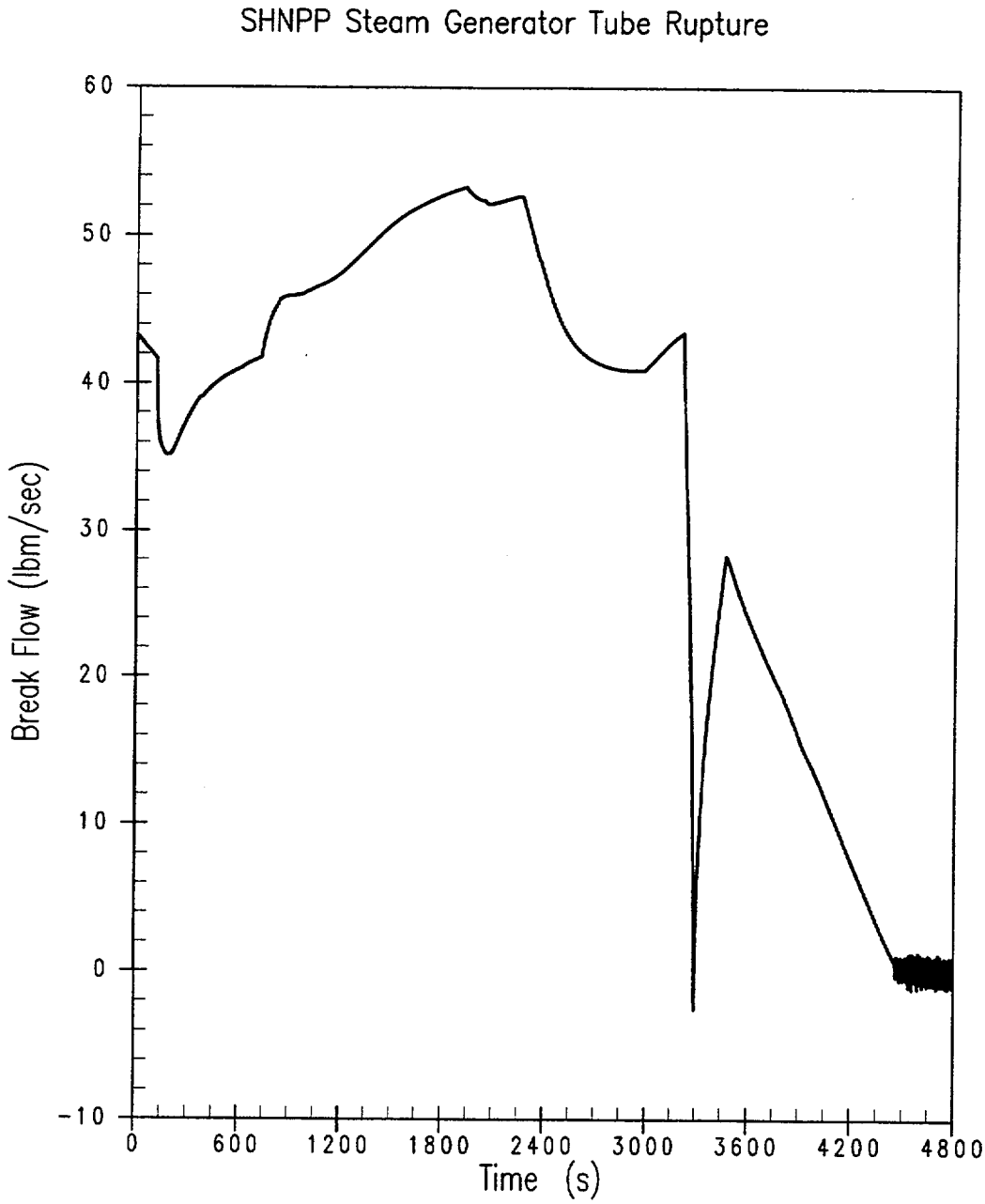

Figure 6.3.2-13
Primary to Secondary Break Flow –
Offsite Radiation Dose Analysis (2787.4 MWt)

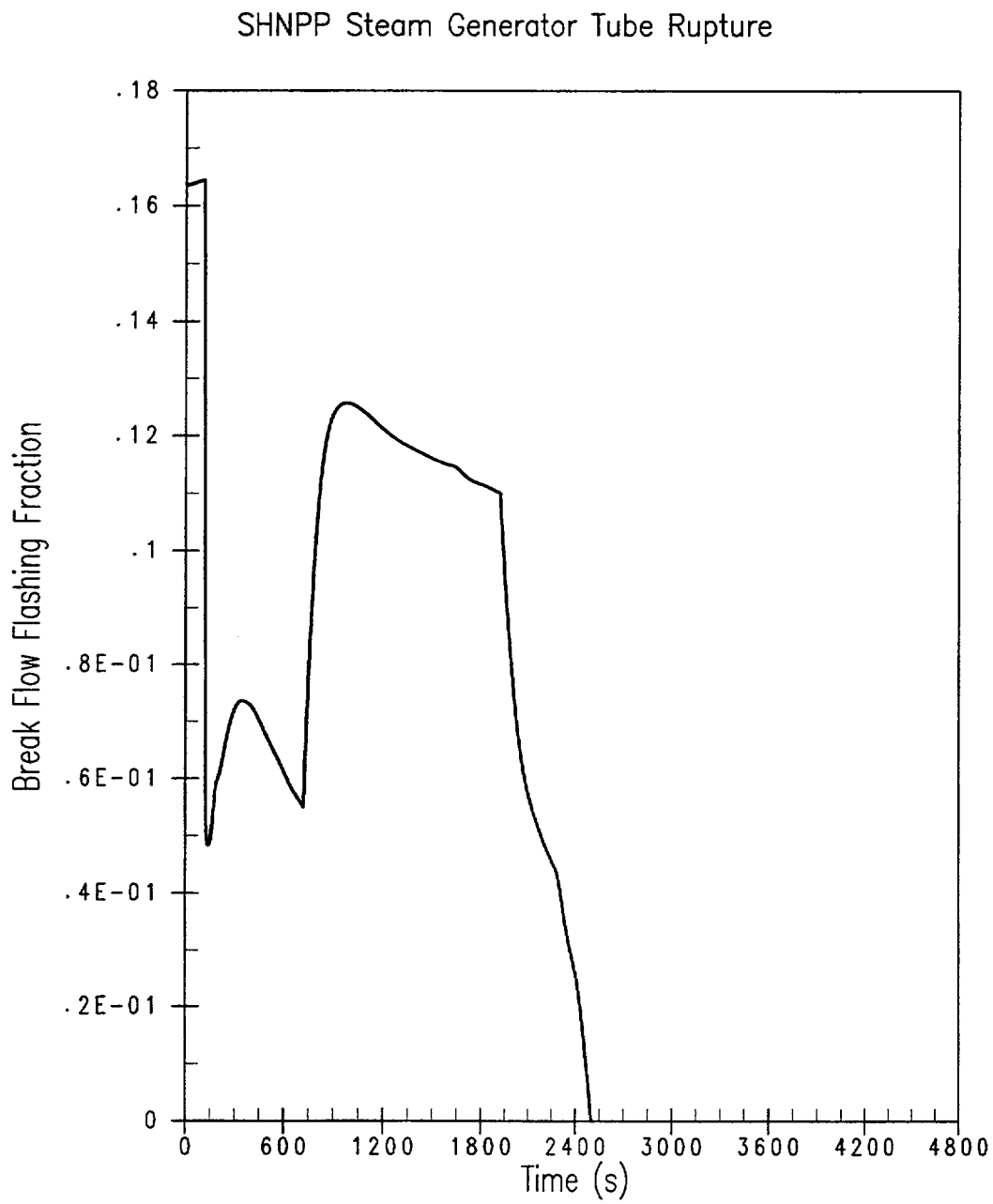

Figure 6.3.2-14
Break Flow Flashing Fraction –
Offsite Radiation Dose Analysis (2787.4 MWt)

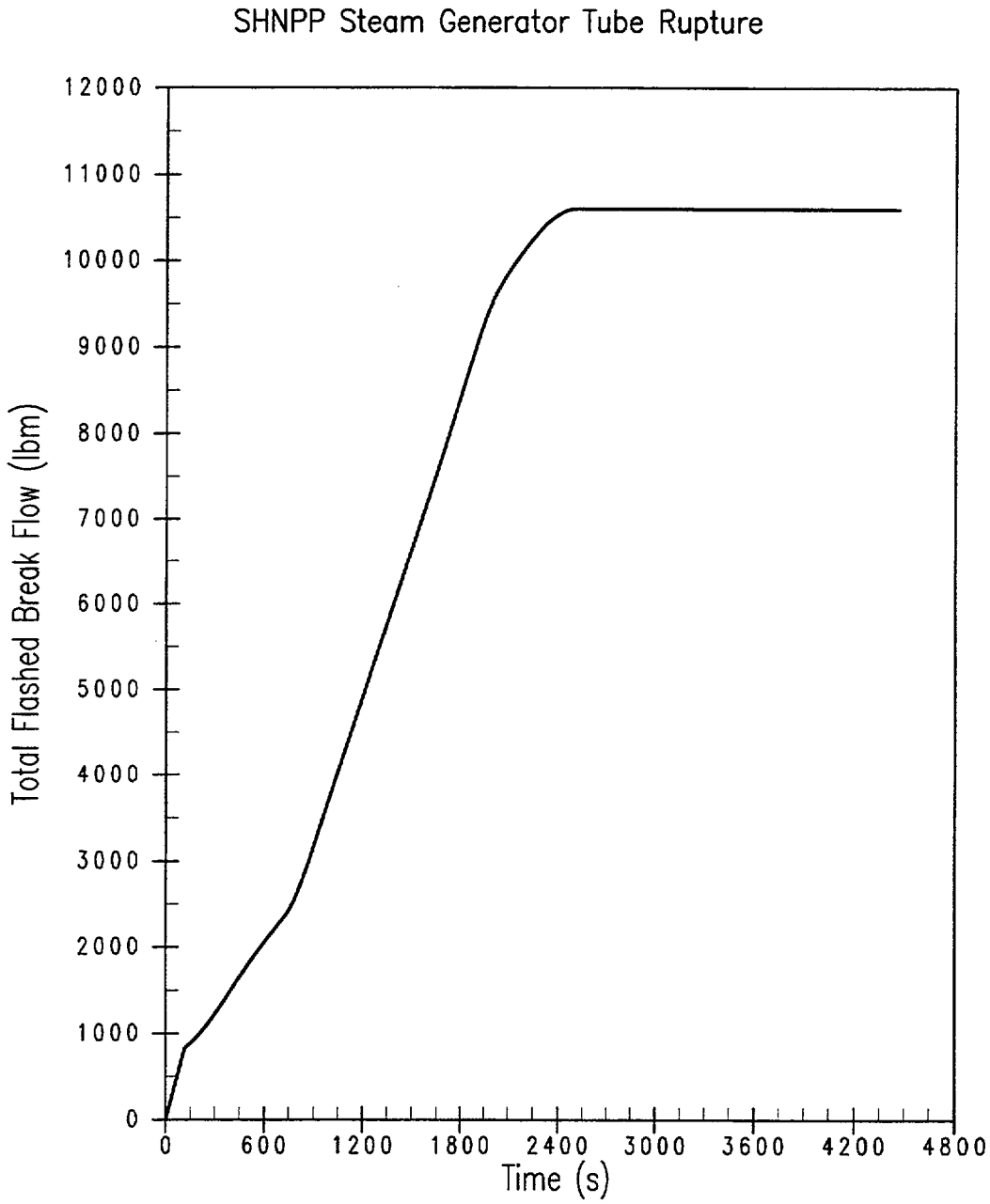

Figure 6.3.2-15
Total Flashed Break Flow –
Offsite Radiation Dose Analysis (2787.4 MWt)

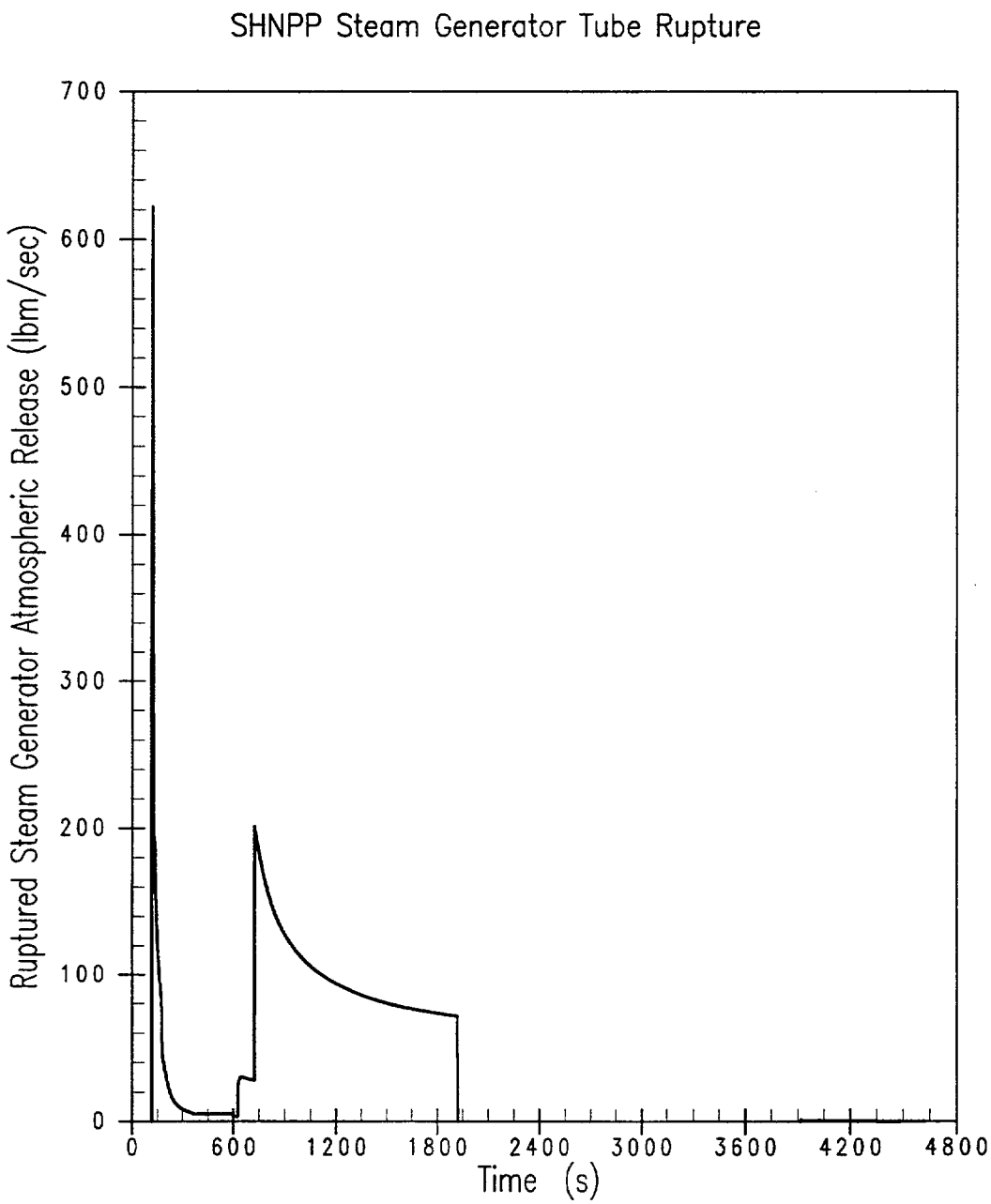

Figure 6.3.2-16
Ruptured SG Mass Release Rate to the Atmosphere – Offsite Radiation Dose Analysis (2787.4 MWt)

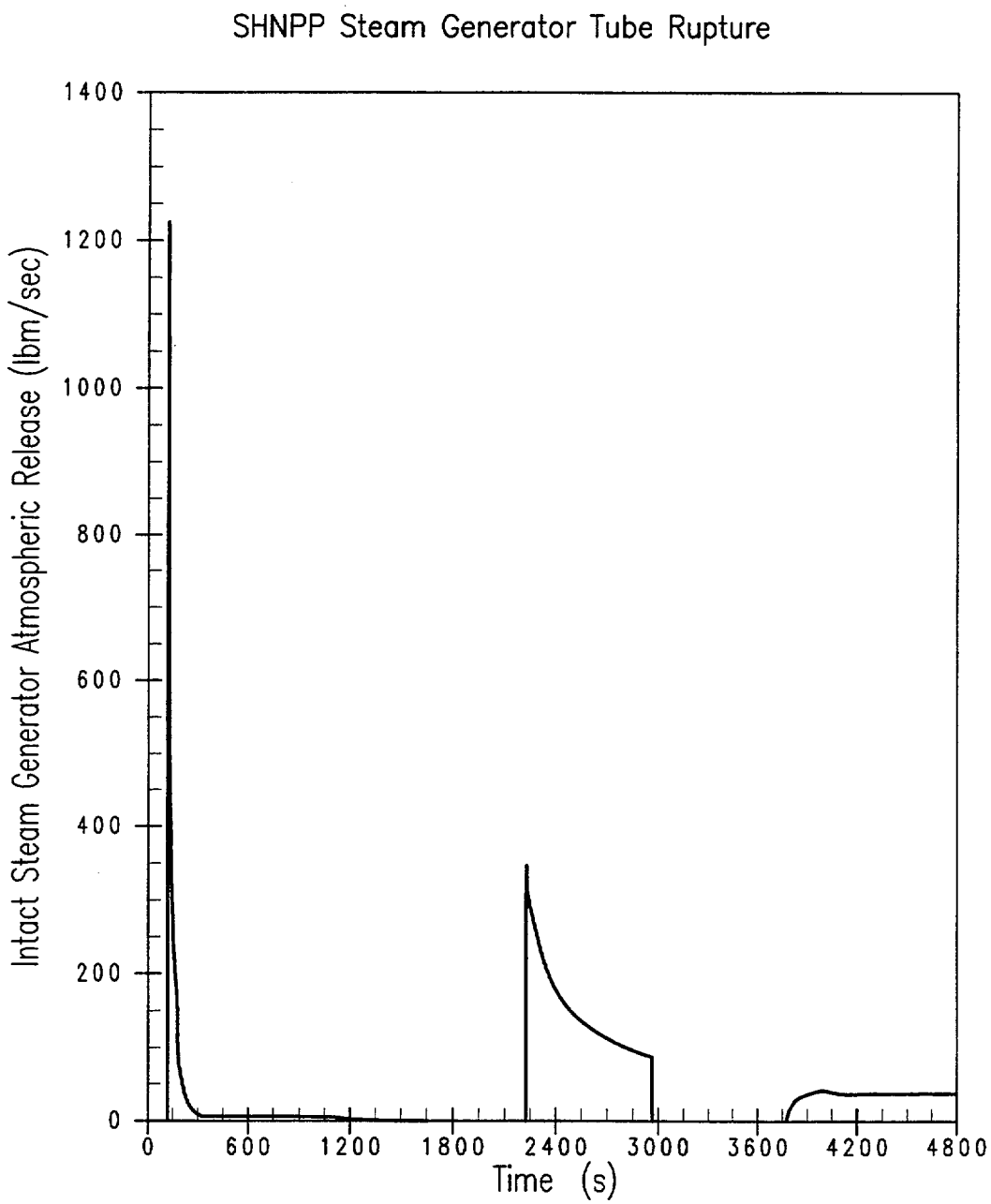

Figure 6.3.2-17
Intact SGs Mass Release Rate to the Atmosphere –
Offsite Radiation Dose Analysis (2787.4 MWt)

6.3.3 Radiological Consequences Analysis

6.3.3.1 Introduction

The evaluation of the radiological consequences of a steam generator tube rupture (SGTR) assumes that the reactor has been operating at the Technical Specification limits for primary coolant activity and primary to secondary leakage for sufficient time to establish equilibrium concentrations of radio-nuclides in the reactor coolant and in the secondary coolant. Radio-nuclides from the primary coolant enter the steam generator, via the ruptured tube and primary to secondary leakage, and are released to the atmosphere through the steam generator safety or power operated relief valves (PORVs) and via the condenser air ejector exhaust.

The quantity of radioactivity released to the environment, due to a SGTR, depends upon primary and secondary coolant activity, iodine spiking effects, primary to secondary break flow, break flow flashing, attenuation of iodine carried by the flashed portion of the break flow, partitioning of iodine between the liquid and steam phases, the mass of fluid released from the generator, and liquid-vapor partitioning in the turbine condenser hot well. All of these parameters were conservatively evaluated for a design basis double ended rupture of a single tube.

The most recent SGTR radiological consequences analysis performed by Westinghouse for HNP, documented in WCAP-12403 and Supplement 1 to WCAP-12403 (References 1 and 2) were performed using the analysis methodology developed in Supplement 1 to WCAP-10698 (Reference 3). The methodology was developed by the SGTR Subgroup of the Westinghouse Owners Group (WOG) and was approved by the Nuclear Regulatory Commission (NRC) in a Safety Evaluation Report (SER) dated December 17, 1985. The SGTR radiological consequences analysis was performed in support of the HNP Model Δ75 replacement steam generator program using this methodology with some variations. These variations in methodology reflect the latest accepted methods and are identified in this report.

Section 6.3.2 of this report presents the mass releases for the SGTR event assuming failure and isolation of the ruptured steam generator PORV for analyses modeling the Model Δ75 replacement steam generators at the uprated NSSS power of 2912.4 MWt and at the current NSSS power of 2787.4 MWt. The resulting offsite and control room doses are calculated in this section.

This section includes the methods and assumptions used to analyze the radiological consequences of the SGTR event, as well as the calculated results.

6.3.3.2 Description of Analyses and Evaluations

Major assumptions and parameters are summarized in Table 6.3.3-1.

6.3.3.2.1 Source Term Assumptions

The radio-nuclide concentrations in the primary and secondary system, prior to and following the SGTR, are determined as follows.

1. The iodine concentrations in the reactor coolant are based upon pre-accident and accident initiated iodine spikes as outlined in the Standard Review Plan (SRP) Section 15.6.3 (Reference 4).

    a. Pre-accident Spike - A reactor transient has occurred prior to the SGTR and has raised the primary coolant iodine concentration to 60 μCi/gm of Dose Equivalent (D.E.) I-131.

    b. Accident-Initiated Spike - The primary coolant iodine concentration is initially at the Technical Specification limit, specified in μCi/gm of D.E. I-131. Following the primary system depressurization and reactor trip associated with the SGTR, an iodine spike is initiated in the primary system. This spike increases the iodine release rate from the fuel to the coolant to a value 500 times greater than the release rate corresponding to the initial primary system iodine concentration. This release rate (the equilibrium iodine appearance rate) is calculated to match the rate of iodine removal from the RCS. Iodine removal from the RCS is the combination of decay, leakage and cleanup.

2. The initial secondary coolant iodine concentration is 0.1 μCi/gm of D.E. I-131.

3. The chemical form of iodine in the primary and secondary coolant is assumed to be elemental.

4. The initial concentration of noble gases in the reactor coolant is based on one-percent defective fuel, which corresponds to the Technical Specification limit of 100/E-bar.

5. No noble gases are present in the secondary system at the start of the event.

The concentration of iodine and noble gas nuclides in the reactor coolant system (RCS) has been calculated based on a one-percent fuel defect level for the SGR/Uprating program. The concentration data presented in Table 6.3.3-2 is used in the SGTR analysis (taken from Section 7.6 of this report). Although this source term data was calculated at the uprated power, it is used in all SGTR dose calculations presented in this report. Since the iodine activity is defined in D.E. I-131, and the distribution of iodine isotopes is not sensitive to small changes in core power, using iodine concentrations based on the uprated power has no impact on the results. The noble gas activity is approximately proportional to the power level assumed in the calculations. Use of the uprated power noble gas activities is conservative and since the noble gas doses are not limiting for a SGTR, they have only a small impact on the results.

The conversion from the one-percent fuel defect values in Table 6.3.3-2 to DE I-131 employs dose conversion factors (DCFs). DCFs are also used in calculating the dose resulting from iodine releases. In the previous analyses the thyroid dose conversion factors are from Regulatory Guide (RG) 1.109 (Reference 5). In order to be consistent with current analysis techniques, thyroid dose conversion factors from International Commission on Radiological Protection (ICRP)-30 (Reference 6) is used in this analysis. These DCFs are used in the calculation of the initial RCS iodine concentrations. The ICRP-30 thyroid dose conversion factors used in the analysis are presented in Table 6.3.3-3.

The current Technical Specification limit for RCS iodine activity is 1.0 μCi/gm D.E. I-131. The HNP Technical Specification limit for RCS iodine activity is being reduced to 0.35 μCi/gm D.E. I-131 for this submittal.

The spike model used in the previous analyses calculated equilibrium iodine appearance rates based on a letdown flow of 60 gpm with 90 percent cleanup. The spike appearance rate is 500 times the equilibrium appearance rate. The conservative spike model calculates the equilibrium iodine appearance rates based on a letdown flow of 120 gpm with perfect cleanup. This flow is conservatively increased by 10 percent to cover uncertainties in the flow. In addition, a total of 42-gpm leakage from the RCS allowed by the Technical Specifications (which also remove iodine from the RCS) is considered in the calculations. The effective letdown flow increases from 54 gpm with the previous non-conservative spike model to 174 gpm with the conservative spike model. The 174 gpm is the total of 120 gpm letdown flow with perfect cleanup increased by 10 percent to 132 gpm (to cover uncertainty), 10 gpm identified leakage from the RCS, 1 gpm unidentified leakage from the RCS, and 31 gpm controlled leakage.

In all cases the spike is allowed to continue until 5 hours from the start of the event. This bounds the time calculated for all iodine initially contained in the gap of the defective fuel to be transferred to the coolant at the spike appearance rate being modeled. In the previous analyses, the spike was assumed to be terminated at 2.78 hours. The spike duration was extended in response to NRC comments on recent analyses performed for other plants. This has little impact on the SGTR analysis, since the majority of the iodine releases end shortly after the ruptured steam generator PORV is isolated at about 30 minutes from the start of the event.

The initial RCS iodine activities used in the analyses are presented in Table 6.3.3-4. The iodine appearance rates used in the analyses are presented in Table 6.3.3-5.

6.3.3.2.2 Dose Calculation Assumptions

Offsite power is assumed to be lost at reactor trip. This assumption was used in the thermal-hydraulic analysis (Section 6.3.2) to maximize break flow and steam release though the ruptured steam generator PORV. Prior to reactor trip, a condenser iodine partition factor of 0.01 is assumed. After reactor trip and loss of offsite power, flow to the condenser is isolated. This

condenser iodine partition factor is consistent with the NUREG-0800 SRP 15.6.3 (Reference 4) steam/water partition coefficient.

The iodine transport model used in this analysis accounts for break flow flashing, steaming, and partitioning. The model assumes that a fraction of the iodine carried by the break flow becomes airborne immediately due to flashing and atomization. The fraction of primary coolant iodine that is not assumed to become airborne immediately mixes with the secondary water and is assumed to become airborne at a rate proportional to the steaming rate. The 0.01 steam/water partition coefficient from NUREG-0800 SRP 15.6.3 (Reference 4) is used. Droplet removal by the dryers is conservatively neglected.

In the iodine transport model, the time dependent iodine removal efficiency for scrubbing of steam bubbles as they rise from the rupture site to the water surface was not calculated and was conservatively neglected. Although this removal was calculated and credited in the previous analyses using a model based on that proposed in NUREG-0409 (Reference 7), it is no longer considered in standard Westinghouse analyses.

Average disintegration energies for nuclides are found in Table 6.3.3-6. These are used in calculating the gamma doses and control room beta skin doses.

All of the iodine in the flashed break flow is assumed to be transferred instantly out of the steam generator to the atmosphere.

The issue of tube bundle uncovery was considered in a Westinghouse Owners Group (WOG) program (Reference 8). The WOG program concluded that the effect of tube uncovery is essentially negligible for the limiting SGTR transient. The WOG program concluded that the steam generator tube uncovery issue could be closed without any further investigation or generic restrictions. The NRC review of the WOG submittal (Reference 9) concluded "... the Westinghouse analyses demonstrate that the effects of partial steam generator tube uncovery on the iodine release for SGTR and non-SGTR events is negligible. Therefore, we agree with your position on this matter and consider this issue resolved." This modeling is different from that used in the previous analyses. Those analyses were completed prior to the resolution of the tube uncovery issue and conservatively modeled the direct release of all iodine transferred to the ruptured steam generator in the break flow when the tubes were assumed to be uncovered.

Since there is no penalty taken for tube uncovery and no iodine scrubbing is credited, the location of the tube rupture is not significant for the radiological analysis. The thermal and hydraulic analysis presented in Section 6.3.2 has conservatively addressed the issue of the location of the tube rupture in the calculations of break flow and flashing of break flow.

No credit is taken for the radioactive decay during release and transport, or for cloud depletion by ground deposition during transport to the control room, site boundary, or outer boundary of the low population zone (LPZ).

All noble gases in the break flow and primary-to-secondary leakage are assumed to be transferred instantly out of the steam generator to the atmosphere.

Iodine and noble gas decay constants are presented in Table 6.3.3-7. These decay constants were calculated from half-lives given in Reference 10.

Short-term atmospheric dispersion factors (χ/Qs) for accident analysis and breathing rates are provided in Table 6.3.3-8. The offsite breathing rates were obtained from NRC RG 1.4 (Reference 11) and the control room breathing rates and occupancy factors are from Murphy-Campe (Reference 12).

Offsite Dose Calculation Model

Thyroid and whole body gamma doses are calculated for 2 hours at the site boundary. At the LPZ thyroid and whole body gamma doses are calculated up to the time all releases are terminated, which is the RHR cut in time used in the thermal and hydraulic analysis.

Offsite thyroid doses are calculated using the following equation.

$$D_{Th} = \sum_i \left[DCF_i \left(\sum_j (IAR)_{ij} (BR)_j (\chi / Q)_j \right) \right]$$

where:

D_{Th} = thyroid dose via inhalation (rem)

DCF_i = thyroid dose conversion factor via inhalation for isotope i (rem/Ci) (Table 6.3.3-3)

$(IAR)_{ij}$ = integrated activity of isotope i released during the time interval j (Ci)

$(BR)_j$ = breathing rate during time interval j (m^3/sec) (Table 6.3.3-8)

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j (sec/m^3) (Table 6.3.3-8)

Offsite whole body gamma doses are calculated using the following equation:

$$D_{WB} = 0.25 \sum_i \left[E_{\gamma i} \left(\sum_j (IAR)_{ij} \ (\chi / Q)_j \right) \right]$$

where:

D_{WB} = whole body dose via cloud immersion (rem)

$E_{\gamma i}$ = average gamma disintegration energy for isotope i (Mev/dis) (Table 6.3.3-6)

$(IAR)_{ij}$ = integrated activity of isotope i released during the time interval j (Ci)

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j (sec/m^3) (Table 6.3.3-8)

The whole body doses are calculated combining the dose from the released noble gases with the dose from the iodine releases. This is more limiting than the previous calculations which only considered the contribution from noble gases. In order to allow a comparison to the previously calculated results, the whole body doses resulting from the noble gases alone are also calculated.

Control Room Dose Calculation Models

Thyroid, whole body gamma, and beta skin doses are calculated for 30 days in the control room. Although all releases are terminated when the RHR system is put in service, the calculation is continued to account for additional doses due to continued occupancy.

The control room is modeled as a discrete volume. The atmospheric dispersion factors calculated for the transfer of activity to the control room intake are used to determine the activity available at the control room intake. The inflow (filtered and unfiltered) to the control room and the control room filtered recirculation flow are used to calculate the concentration of activity in the control room. Control room parameters used in the analysis are presented in Table 6.3.3-9.

Control room thyroid doses are calculated using the following equation:

$$D_{Th} = \sum_i \left[DCF_i \left(\sum_j Conc_{ij} * (BR)_j \right) \right]$$

where:

D_{Th} = thyroid dose via inhalation (rem)

DCF_i = thyroid dose conversion factor via inhalation for isotope i (rem/Ci) (Table 6.3.3-3)

$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)

$(BR)_j$ = breathing rate during time interval j (m^3/sec) (Table 6.3.3-8)

Control room whole body doses are calculated using the following equation:

$$D_{WB} = 0.25 * \left(\frac{1}{GF}\right) * \sum_i E_{\gamma i} \left(\sum_j Conc_{ij}\right)$$

where:

D_{WB} = whole body dose via cloud immersion in rem

GF = geometry factor, calculated based on Reference 12, using the equation

$$GF = \frac{1173}{V^{0.338}}$$ where V is the control room volume in ft^3

$E_{\gamma i}$ = average gamma disintegration energy for isotope i (Mev/dis) (Table 6.3.3-6)

$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)

Control room skin doses are calculated using the following equation:

$$D_\beta = 0.23 * \sum_i E_{\beta i} \left(\sum_j Conc_{ij}\right)$$

where:

D_β = whole body dose via cloud immersion (rem)

$E_{\beta i}$ = average beta disintegration energy for isotope i (Mev/dis) (Table 6.3.3-6)

$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)

6.3.3.2.3 Mass Transfer Assumptions

Break flow, flashing break flow and steam releases from the intact and ruptured steam generators are modeled using data from the thermal and hydraulic analysis in Section 6.3.2 of this report.

A total primary to secondary leak rate is assumed to be 1.0 gpm. The leak is assumed to be distributed with 0.7 gpm to the two intact steam generators and 0.3 gpm to the ruptured steam generator. The leakage to the intact steam generators is assumed to persist for the duration of the accident. This modeling is consistent with the previous analyses. Atmospheric conditions are assumed in determining the density for this leakage.

In addition to the releases calculated in the thermal hydraulic analysis presented in Section 6.3.2, steam released from the ruptured steam generator to the turbine driven auxiliary feedwater (TDAFW) pump is considered in the dose analysis. A flow of 41,310 lbm/hr is considered from the time of auxiliary feedwater initiation until the ruptured steam generator is isolated. The iodine contained in this steam, determined from the steam generator activity and the water/steam partition coefficient of 100, is assumed to be released directly to the atmosphere. This flow was not modeled in the previous analyses and is not required by the Reference 3 methodology. Inclusion of this flow is conservative since it is assumed to be released directly to the atmosphere.

6.3.3.3 Description of Analyses Cases

Offsite and control room doses are calculated for the two thermal hydraulic analyses presented in Section 6.3.2 of this report. One set of dose calculations corresponds to the analysis performed at the uprated NSSS power of 2912.4 MWt with Model Δ75 replacement steam generators. For this set the mass transfer data is taken from Table 6.3.2-2 and Figures 6.3.2-6, 6.3.2-8, 6.3.2-9, and 6.3.2-10. A second set of dose calculations corresponds to the analysis performed at the current NSSS power of 2787.4 MWt with Model Δ75 replacement steam generators. For this set the mass transfer data is taken from Table 6.3.2-3 and Figures 6.3.2-13, 6.3.2-15, 6.3.2-16, and 6.3.2-17.

Each set of calculations determines the thyroid doses based on a pre-accident iodine spike of 60.0 μCi/gm D.E. I-131 primary coolant activity. Thyroid doses for two assumed accident-initiated iodine spike appearance rates are calculated for each set of thermal hydraulic results, as described in Section 6.3.3.2.1. Both spike assumptions consider 0.1 μCi/gm D.E. I-131 secondary activity.

The whole body doses are calculated combining the dose from the released noble gases with the dose from the iodine releases, as described in Section 6.3.3.2.2. In the Reference 1 and 2 analyses the whole body doses reported were based solely on the noble gas contribution consistent with the Reference 3 methodology. The current industry practice is to include the iodine contribution in the whole body doses. The whole body doses are calculated with the limiting iodine releases (either pre-accident spike or accident-initiated iodine spike).

6.3.3.4 Acceptance Criteria

The acceptance limits for doses must be satisfied for a SGTP. The offsite dose limits are specified in NUREG-0800 SRP 15.6.3 (Reference 4). The doses at the site boundary (SB) and the LPZ for a SGTR with an assumed pre-accident iodine spike must be within the 10 CFR 100 limits, (i.e., less than 300 rem thyroid and 25 rem whole body). The doses at the SB and the LPZ for a SGTR with an assumed accident-initiated iodine spike must be within a small fraction (10 percent) of the 10 CFR 100 limits, i.e., less than 30-rem thyroid and 2.5-rem whole body. The control room dose limits are specified in NUREG-0800 SRP 6.4 (Reference 13) based on General Design Criteria (GDC) 19. Doses in the control room must be less than 30-rem thyroid, 5-rem whole body, and 30-rem beta-skin.

The site boundary doses are calculated for 2 hours. The LPZ doses are calculated up to the time all releases are terminated, which is the Residual Heat Removal (RHR) cut in time (8 hours) used in the thermal and hydraulic analysis in Section 6.3.2. The control room doses are calculated for 30 days.

6.3.3.5 Results

The pre-accident iodine spike thyroid doses for the SGTR analysis with Model Δ75 replacement steam generators at the uprated NSSS power of 2912.4 MWt and at the current NSSS power level of 2787.4 MWt are tabulated in Table 6.3.3-10. The table includes the most recent Westinghouse reported doses (from Reference 2) and the applicable limit. The results in the table demonstrate that the SGR/Uprating does not result in an increase in the previous pre-accident iodine spike thyroid doses. The applicable limits are met.

Table 6.3.3-11 presents the accident-initiated iodine spike doses calculated based on a primary coolant iodine limit of 0.35 μCi/gm D.E. I-131, and spike appearance rates calculated with conservative assumptions. The results in the table demonstrate that the applicable limits are met. The reduction in allowable primary coolant iodine activity to 0.35 is sufficient to offset the penalty associated with the revised spike appearance rate calculations.

Table 6.3.3-12 presents the whole body doses calculated using only noble gas releases, consistent with those reported in Reference 2. The table includes the most recent Westinghouse reported doses (from Reference 2) and the applicable limit. The results in the table demonstrate that the SGR/Uprating does not result in an increase in the previous whole body doses. These results

provide a direct comparison to the current analysis of record. Table 6.3.3-13 presents the whole body doses calculated including the contribution from iodines. The iodine contribution from the limiting iodine spike case, determined to be the pre-accident spike, is used. The results in the table demonstrate that the applicable limits are met.

Table 6.3.3-14 presents the control room beta skin doses, including the allowable guideline value. The iodine contribution from the limiting iodine spike case is used. The results in the table demonstrate that the applicable limits are met.

6.3.3.6 Conclusions

The potential radiological consequences of a SGTR were evaluated for HNP in support of the SGR/Uprating program. Since it was analyzed in Section 6.3.1 that steam generator overfill will not occur for a design basis SGTR, an analysis was performed to determine the offsite radiation doses assuming the limiting single failure for offsite doses. The thermal hydraulic results from this analysis are presented in Section 6.3.2. The resulting doses at the exclusion area boundary, low population zone, and control room (presented in Section 6.3.3) are within the allowable guidelines. The analysis also demonstrated that the doses do not increase, relative to previously reported values (Reference 2), as a result of the SGR/Uprating.

6.3.3.7 References

1. WCAP-12403, "LOFTTR2 Analysis for a Steam Generator Tube Rupture with Revised Operator Action Times for Shearon Harris Nuclear Power Plant," Huang, Lewis, Marmo, Rubin, November 1989.

2. Supplement 1 to WCAP-12403, "Steam Generator Tube Rupture Analysis for Shearon Harris Nuclear Power Plant," Lewis, Lowe, Monahan, Rubin, Tanz, November 1992.

3. Supplement 1 to WCAP-10698-P-A, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," Lewis, Huang, Rubin, March 1986.

4. NUREG-0800, Standard Review Plan Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure (PWR)," Rev. 2, July 1981.

5. Regulatory Guide 1.109, Rev. 1, "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," US Nuclear Regulatory Commission, October 1977.

6. International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, Volume 3 No. 1-4, 1979.

7. NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture," A. K. Postma, P. S. Tam.

8. WCAP-13247, "Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue," March 1992.

9. Letter from Robert C. Jones to Lawrence A. Walsh, "Westinghouse Owners Group-Steam Generator Tube Uncovery Issue," March 10, 1993.

10. ENDF-223, "ENDF/B-IV Fission-Product Files: Summary of Major Nuclide Data," T. R. England and R. E. Schenter, October 1975.

11. NRC Regulatory Guide 1.4, Revision 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for Pressurized Water Reactors," June 1974.

12. K. G. Murphy and K. W. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," published in Proceedings of 13th AEC Air Cleaning Conference, Atomic Energy Commission (now NRC), August 1974.

13. NUREG-0800, Standard Review Plan 6.4, "Control Room Habitability System," Revision 2, July 1981.

Table 6.3.3-1 Summary of Parameters Used in Evaluating the Radiological Consequences of a Steam Generator Tube Rupture	
I. Source Data	
A. Core Power Level, MW_t	2900
B. Reactor Coolant Iodine Activity:	
1. Accident-Initiated Spike	The initial RC iodine activities are presented in Table 6.3.3-4. The iodine appearance rates assumed for the accident-initiated spike are presented in Table 6.3.3-5.
2. Pre-Accident Spike	Primary coolant iodine activities based on 60 μCi/gm of D.E. I-131 are presented in Table 6.3.3-4.
C. Noble Gas Activity	Primary coolant noble gas activities based on 1-percent fuel defects are presented in Table 6.3.3-2. No noble gases are contained in the secondary system.
D. Secondary System Initial Activity	Dose equivalent of 0.1 μCi/gm of I-131, presented in Table 6.3.3-4.
E. Reactor Coolant Initial Mass, grams	
Uprated Power	1.73×10^8
Current Power	1.73×10^8
F. Steam Generator Initial Mass (each), grams	
Uprated Power	4.34×10^7
Current Power	4.38×10^7
G. Offsite power	Lost at time of reactor trip
H. Primary-to-Secondary Leakage Duration for Intact SG, hours	8
I. Species of Iodine	100 percent elemental

Table 6.3.3-1 (Cont.) Summary of Parameters Used in Evaluating the Radiological Consequences of a Steam Generator Tube Rupture	
II. Activity Release Data	
A. Ruptured Steam Generator	
1. Rupture Flow	
Uprated Power	See Table 6.3.2-2 & Figure 6.3.2-6
Current Power	See Table 6.3.2-3 & Figure 6.3.2-13
2. Flashed Rupture Flow	
Uprated Power	See Table 6.3.2-2 & Figure 6.3.2-8
Current Power	See Table 6.3.2-3 & Figure 6.3.2-15
3. Steam Releases	
Uprated Power	See Table 6.3.2-2 & Figure 6.3.2-9
Current Power	See Table 6.3.2-3 & Figure 6.3.2-16
	In both cases an additional 41,310 lbm/hr to TDAFW pump is modeled until ruptured SG isolation.
4. Iodine Partition Factor for Rupture Flow	
Non-flashed	100
Flashed	1.0
B. Intact Steam Generators	
1. Primary-to-Secondary Leakage, gpm	0.7
2. Steam Releases	
Uprated Power	See Table 6.3.2-2 & Figure 6.3.2-10
Current Power	See Table 6.3.2-3 & Figure 6.3.2-17
3. Iodine Partition Factor	100
C. Condenser	
1. Iodine Partition Factor	100
D. Atmospheric Dispersion Factors	See Table 6.3.3-8

Table 6.3.3-2 Reactor Coolant Fission Product Specific Activity Based on 1-Percent Fuel Defects	
Nuclide	**Specific Activity (μCi/gm)**
I-131	1.71E+00
I-132	2.47E+00
I-133	7.23E+00
I-134	5.67E-01
I-135	1.84E+00
Kr-85m	1.7
Kr-85	10.6
Kr-87	1.1
Kr-88	3.2
Xe-131m	3.4
Xe-133m	4.9
Xe-133	276.4
Xe-135m	0.4
Xe-135	8.5
Xe-138	0.6

Table 6.3.3-3 Thyroid Dose Conversion Factors (Reference 6*)	
Nuclide	**DCF (Rem/Curie)**
I-131	1.07×10^6
I-132	6.29×10^3
I-133	1.81×10^5
I-134	1.07×10^3
I-135	3.14×10^4

*Reference 6 provides the dose conversion factors in units of sievert/becquerel.

Table 6.3.3-4 Iodine Specific Activities (μCi/gm) in the Primary Coolant Based on 0.35 and 60.0 μCi/gm of D.E. I-131 and in the Secondary Coolant Based on 0.1 μCi/gm of D.E. I-131			
	Primary Coolant		**Secondary Coolant**
Nuclide	**0.35 μCi/gm**	**60 μCi/gm**	**0.1 μCi/gm**
1-131	0.200	34.20	0.0570
1-132	0.288	49.38	0.0823
1-133	0.843	144.48	0.2408
1-134	0.066	11.34	0.0189
1-135	0.215	36.78	0.0613

Table 6.3.3-5 Iodine Spike Appearance Rates (Curies/Minute) Based on 0.35 μCi/gm of D.E. I-131 Primary Coolant Activity						
Primary Activity	**Assumptions**	**I-131**	**I-132**	**I-133**	**I-134**	**I-135**
0.35 μCi/gm D.E. I-131	Conservative	67.0	220.5	318.0	97.0	103.5

Table 6.3.3-6 Disintegration Energies (Reference 10)		
Nuclide	**Gamma Disintegration Energy (Mev/Dis)**	**Beta Disintegration Energy (Mev/Dis)**
I-131	0.38	0.19
I-132	2.2	0.52
I-133	0.6	0.42
I-134	2.6	0.69
I-135	1.4	0.43
Kr-85m	0.16	0.25
Kr-85	0.0023	0.25
Kr-87	0.79	1.3
Kr-88	2.2	0.25
Xe-131m	0.0029	0.16
Xe-133m	0.02	0.21
Xe-133	0.03	0.15
Xe-135m	0.43	0.099
Xe-135	0.25	0.32
Xe-138	1.2	0.66

Table 6.3.3-7 Decay Constants (Reference 10)	
Nuclide	**Decay Constant (1/hr)**
I-131	0.00359
I-132	0.303
I-133	0.0333
I-134	0.791
I-135	0.105
Kr-85m	0.155
Kr-85	7.37E-6
Kr-87	0.547
Kr-88	0.248
Xe-131m	0.00241
Xe-133m	0.0130
Xe-133	0.00546
Xe-135m	2.72
Xe-135	0.0756
Xe-138	2.93

Table 6.3.3-8
Atmospheric Dispersion Factors and Breathing Rates

Time (hours)	Exclusion Area Boundary χ/Q (sec/m^3)	Low Population Zone χ/Q (sec/m^3)	Control Room χ/Q (sec/m^3)	Offsite Breathing Rate (m^3/sec)	Control Room Breathing Rate (m^3/sec)	Control Room Occupancy Factor
0 - 2	6.17×10^{-4}	1.4×10^{-4}	4.08×10^{-3}	3.47×10^{-4}	3.47×10^{-4}	1.0
2 - 8	----	1.4×10^{-4}	4.08×10^{-3}	3.47×10^{-4}	3.47×10^{-4}	1.0
8 - 24	----	----	1.16×10^{-3}	1.75×10^{-4}	3.47×10^{-4}	1.0
24 - 96	----	----	3.25×10^{-4}	2.32×10^{-4}	3.47×10^{-4}	0.6
> 96	----	----	1.23×10^{-5}	2.32×10^{-4}	3.47×10^{-4}	0.4

Table 6.3.3-9 Control Room Model	
Control Room Isolation Signal Generated	Time of SI signal from Section 6.3.2
Delay in Control Room Isolation After Isolation Signal is Generated	30 Seconds
Control Room Volume	71000 ft^3
Control Room HVAC Ductwork Volume	2704 ft^3
Control Room Unfiltered In-Leakage	80 cfm
Control Room Unfiltered Inflow	
Normal Mode	1050 cfm
Emergency Mode	0 cfm
Control Room Filtered Inflow	
Normal Mode	0 cfm
Emergency Mode	400 cfm
Control Room Filtered Recirculation	
Normal Mode	0 cfm
Emergency Mode	3600 cfm
Control Room Filter Efficiency	99%

Table 6.3.3-10 Pre-Accident Iodine Spike Thyroid Doses				
	RSG and Uprated Doses (Rem)	**RSG with Current Power Doses (Rem)**	**Reference 2 Doses (Rem)**	**Allowable Guideline Value**
Pre-Accident Iodine Spike - Thyroid				
Exclusion Area Boundary (0-2 hr.)	57.74	56.47	90.4	300
Low Population Zone (0-8 hr.)	13.84	13.51	20.7	300
Control Room (0-30 Days)	10.93	10.43	Not Calculated	30

Table 6.3.3-11 Accident-Initiated Iodine Spike Thyroid Doses Iodine Spike Appearance Rates Based on 0.35 μCi/gm of D.E. I-131 Primary Coolant Activity			
	RSG and Uprated Doses (Rem)	**RSG with Current Power Doses (Rem)**	**Allowable Guideline Value**
Accident-Initiated Iodine Spike - Thyroid			
Exclusion Area Boundary (0-2 hr.)	15.86	15.58	30
Low Population Zone (0-8 hr.)	4.07	3.97	30
Control Room (0-30 Days)	2.54	2.48	30

Table 6.3.3-12 Whole Body Doses From Noble Gases				
	RSG and Uprated Doses (Rem)	**RSG with Current Power Doses (Rem)**	**Reference 2 Doses (Rem)**	**Allowable Guideline Value**
Whole Body Gamma Dose				
Exclusion Area Boundary (0-2 hr.)	0.170	0.169	0.2	2.5
Low Population Zone (0-8 hr.)	0.039	0.038	0.1	2.5
Control Room (0-30 Days)	0.036	0.036	Not Calculated	5

Table 6.3.3-13 Total Whole Body Doses			
	RSG and Uprated Doses (Rem)	**RSG with Current Power Doses (Rem)**	**Allowable Guideline Value**
Whole Body Gamma Dose			
Exclusion Area Boundary (0-2 hr.)	0.35	0.34	2.5
Low Population Zone (0-8 hr.)	0.08	0.08	2.5
Control Room (0-30 Days)	0.04	0.04	5

Table 6.3.3-14 Control Room Skin Doses				
	RSG and Uprated Doses (Rem)	**RSG with Current Power Doses (Rem)**	**Reference 2 Doses (Rem)**	**Allowable Guideline Value**
Beta Skin Dose				
Control Room (0-30 Days)	3.04	3.01	Not Calculated	30

6.4 LOCA Mass and Energy Releases

The uncontrolled release of pressurized high temperature reactor coolant, or loss-of-coolant accident (LOCA), results in release of steam and water into the containment. This, in turn, results in increases in the local subcompartment pressures, and an increase in the global containment pressure and temperature. There are both long and short-term issues relative to a postulated LOCA that must be considered for the SGR/Uprating for the Harris Nuclear Plant (HNP).

The long-term LOCA mass and energy (M&E) releases, addressed in Section 6.4.1, are utilized as input to the containment integrity analysis, which demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a hypothetical large break LOCA.

The short-term LOCA-related M&E releases, addressed in Section 6.4.2, are used as input to the subcompartment analyses, which are performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) accompanying a high energy line pipe rupture within that subcompartment.

6.4.1 Long-Term LOCA M&E Releases

6.4.1.1 Introduction

The limiting long-term LOCA mass and energy releases are analyzed to approximately $3x10^7$ seconds, or one year, and are utilized as input to the containment integrity analysis. The containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure and limiting the temperature excursion to less than the Environmental Qualification (EQ) acceptance limits. The containment safeguards systems must also be capable of limiting the peak containment pressure to less than the Integrated Leak Rate Test (ILRT) pressure and reducing the pressure to less than 50 percent of the calculated pressure in 24 hours. For the SGR/Uprating program, Westinghouse generated the M&E releases using the March 1979 model, described in Reference 1, which includes the NRC review and approval letter. This methodology has previously been applied to the HNP (Reference 2). The long-term LOCA M&E releases generated by Westinghouse for this program have been provided to CP&L for use in the containment integrity analysis and EQ reviews. (See the BOP Licensing Report.)

Section 6.4.1 addresses the long-term LOCA M&E releases for the hypothetical double-ended pump suction (DEPS) rupture and double-ended hot-leg (DEHL) rupture break cases.

The mass and energy release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. For example, the Reactor Coolant System (RCS) operating temperatures are chosen to bound the highest average coolant

System (RCS) operating temperatures are chosen to bound the highest average coolant temperature range of all operating cases, and a temperature uncertainty allowance of (+6.0°F) is then added. Nominal parameters are used in certain instances. For example, the RCS pressure in this analysis is based on a nominal value of 2250 psia plus an uncertainty allowance (+51 psi).

All input parameters are chosen consistent with accepted analysis methodology. Some of the most-critical items are the RCS initial conditions, core decay heat, safety injection flow, and primary and secondary metal mass and steam generator heat release modeling. Specific assumptions concerning each of these items are discussed next. Tables 6.4.1-1, 6.4.1-2 and 6.4.1-3 present key data assumed in the analysis.

The core rated power of 2900 MWt adjusted for calorimetric error (+2 percent of power) was used in the analysis. As previously noted, the use of RCS operating temperatures to bound the highest average coolant temperature range were used as bounding analysis conditions. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures that are at the maximum levels attained in steady state operation. Additionally, an allowance to account for instrument error and deadband is reflected in the initial RCS temperatures. The selection of 2250 psia as the limiting pressure is considered to affect the blowdown phase results only, since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally the RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Thus, 2250 psia plus uncertainty was selected for the initial pressure as the limiting case for the long-term M&E release calculations.

The selection of the fuel design features for the long-term M&E release calculation is based on the need to conservatively maximize the energy stored in the fuel at the beginning of the postulated accident (i.e., to maximize the core stored energy). The margin in core-stored energy was chosen to be +15 percent. Thus, the analysis very conservatively accounts for the stored energy in the core.

Margin in RCS volume of 3 percent (1.6 percent allowance for thermal expansion and 1.4 percent for uncertainty) is modeled.

A uniform steam generator tube plugging (SGTP) level of 0 percent is modeled. This assumption maximizes the reactor coolant volume and fluid release by considering the RCS fluid in all SG tubes. During the post-blowdown period the steam generators are active heat sources, as significant energy remains in the secondary metal and secondary mass that has the potential to be transferred to the primary side. The 0-percent SGTP assumption maximizes heat transfer area and therefore, the transfer of secondary heat across the SG tubes. Additionally, this assumption

reduces the reactor coolant loop resistance, which reduces the pressure drop upstream of the break for the pump suction breaks and increases break flow. Thus, the analysis very conservatively accounts for the level of SGTP.

Regarding safety injection flow, the M&E release calculation considered configurations/failures to conservatively bound respective alignments. These cases include (1) a Minimum Safeguards case (one Charging/Safety Injection pump [CH/SI] and one Low Head Safety Injection [LHSI] pump) and (2) a Maximum Safeguards case (two CH/SI and two LHSI pumps).

The following assumptions were employed to ensure that the M&E releases are conservatively calculated, thereby maximizing energy release to containment:

1. Maximum expected operating temperature of the RCS (100-percent full-power conditions)

2. Allowance for RCS temperature uncertainty (+6.0°F)

3. Margin in RCS volume of 3 percent (which is composed of 1.6-percent allowance for thermal expansion, and 1.4 percent for uncertainty)

4. Core rated power of 2900 MWt

5. Allowance for calorimetric error (+2 percent of power)

6. Conservative heat transfer coefficients (i.e., steam generator primary/secondary heat transfer and reactor coolant system metal heat transfer)

7. Allowance in core-stored energy for effect of fuel densification

8. A margin in core-stored energy (+15 percent to account for manufacturing tolerances)

9. An allowance for RCS initial pressure uncertainty (+51 psi)

10. A maximum containment backpressure equal to design pressure (45 psig)

11. Allowance for RCS flow uncertainty (-2.1 percent)

12. SGTP leveling (0-percent uniform)

    - Maximizes reactor coolant volume and fluid release
    - Maximizes heat transfer area across the SG tubes
    - Reduces coolant loop resistance, which reduces the ΔP upstream of the break for the pump suction breaks and increases break flow

Thus, based on the previously discussed conditions and assumptions, a bounding analysis for the HNP was made for the release of M&E from the RCS in the event of a LOCA at 2900 MWt.

6.4.1.2 Description of Analyses and Evaluations

The evaluation model used for the long-term LOCA M&E release calculations is the March 1979 model described in Reference 1. This evaluation model has been reviewed and approved generically by the Nuclear Regulatory Commission (NRC). The approval letter is included with Reference 1. This methodology has previously been applied to the HNP (Reference 2).

6.4.1.2.1 LOCA Mass and Energy Release Phases

The containment system receives mass and energy releases following a postulated rupture in the RCS. These releases continue over a time period, which, for the LOCA M&E analysis, is typically divided into four phases.

1. Blowdown – the period of time from accident initiation (when the reactor is at steady state operation) to the time that the RCS and containment reach an equilibrium state.

2. Refill – the period of time when the lower plenum is being filled by accumulator and Emergency Core Cooling System (ECCS) water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment M&E releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of M&E to containment. Thus, the refill period is conservatively neglected in the M&E release calculation.

3. Reflood – begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Post-reflood (Froth) – describes the period following the reflood phase. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators prior to exiting the break as steam. After the broken loop steam generator cools, the break flow becomes two-phase.

6.4.1.2.2 Computer Codes

The Reference 1 mass and energy release evaluation model is comprised of M&E release versions of the following codes: SATAN VI, WREFLOOD, FROTH, and EPITOME. These codes were used to calculate the long-term LOCA M&E releases for the HNP SGR/Uprating program. These codes have been used for this analysis since the original plant licensing.

SATAN VI calculates blowdown, the first portion of the thermal-hydraulic transient following break initiation, including pressure, enthalpy, density, M&E flowrates, and energy transfer between primary and secondary systems as a function of time.

The WREFLOOD code addresses the portion of the LOCA transient where the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and when water supplied by the ECCS refills the reactor vessel and provides cooling to the core. The most important feature of WREFLOOD is the steam/water mixing model, discussed in subsection 6.4.1.4.2.

FROTH models the post-reflood portion of the transient. The FROTH code is used for the steam generator heat addition calculation from the broken and intact loop steam generators.

EPITOME continues the FROTH post-reflood portion of the transient from the time at which the secondary equilibrates to containment design pressure to the end of the transient. It also compiles a summary of data on the entire transient, including formal instantaneous M&E release tables and M&E balance tables with data at critical times.

6.4.1.2.3 Break Size and Location

Generic studies (Reference 1, Section 3) have been performed with respect to the effect of postulated break size on the LOCA M&E releases. The double-ended guillotine break (DEGB) has been found to be limiting due to larger mass flowrates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system loop can be postulated for pipe rupture for any release purposes:

1. Hot leg (between vessel and steam generator)
2. Cold leg (between pump and vessel)
3. Pump suction (between steam generator and pump)

The break locations analyzed for this program are the DEPS rupture (10.48 ft^2) and the DEHL rupture (9.18 ft^2). Break M&E releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA for the DEPS cases. For the DEHL case, the releases were calculated only for the blowdown. The following information provides a discussion of each break location.

- The DEHL rupture has been shown in previous studies (Reference 1, Section 3.1) to result in the highest blowdown M&E release rates. Although the core flooding rate would be the highest for this break location, the amount of energy released from the steam generator secondary is minimal because the majority of fluid that exits the core vents directly to

containment, bypassing the steam generators. As a result, the reflood M&E releases are reduced significantly as compared to either the pump suction, or cold-leg break locations where the core exit mixture must pass through the steam generators before venting through the break. For the hot-leg break, generic studies have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the containment pressure would continually decrease). Therefore, only the M&E releases for the hot-leg break blowdown phase are calculated and presented in subsection 6.4.1.4 of this report.

- The cold-leg break location has also been found in previous studies (Reference 1, Section 3.1) to be much less limiting in terms of the overall containment energy releases. The cold-leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that for the pump suction break. During reflood, the flooding rate is greatly reduced, and the energy release rate into the containment is reduced. Therefore, the cold-leg break is bounded by other breaks and no further evaluation is necessary.

- The pump suction break combines the effects of the relatively high core-flooding rate, as in the hot-leg break, and the additional stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the RCS in calculating the releases to containment.

6.4.1.2.4 Application of Single-Failure Criterion

An inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the safety injection system. This is not an issue for the blowdown period, which is limited by the DEHL break, since the combination of signal delay plus diesel delay and additional delays in starting the ECCS pumps results in an SI delivery time after the end of blowdown.

Generally, two cases are analyzed to assess the effects of a single failure. The first case assumes minimum safeguards SI flow based on the postulated single failure of an emergency diesel generator. This results in the loss of one train of safeguards equipment. The other case assumes maximum safeguards SI flow based on no postulated failures that would impact the amount of ECCS flow.

6.4.1.3 Acceptance Criteria

A large LOCA is classified as an ANS Condition IV event, an infrequent fault. To satisfy the NRC acceptance criteria presented in the Standard Review Plan Section 6.2.1.3, the relevant requirements are as follows:

- 10 CFR 50, Appendix A
- 10 CFR 50, Appendix K, paragraph I.A

In order to meet these requirements, the following must be addressed:

- Sources of energy
- Break size and location
- Calculation of each phase of the accident

6.4.1.4 Results

6.4.1.4.1 Blowdown Mass and Energy Release Data

The SATAN-VI code is used for computing the blowdown transient. The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform, and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is incorporated into the analysis. The methodology for the use of this model is described in Reference 1.

Table 6.4.1-4 presents the calculated mass and energy release for the blowdown phase of the DEHL break. For the hot-leg break M&E release tables, break path 1 refers to the M&E exiting from the reactor vessel side of the break; and break path 2 refers to the M&E exiting from the steam generator side of the break.

Table 6.4.1-5 presents the calculated M&E releases for the blowdown phase of the DEPS break with either minimum or maximum ECCS flows. For the pump suction breaks, break path 1 in the M&E release tables refers to the M&E exiting from the steam generator side of the break; break path 2 refers to the M&E exiting from the pump side of the break.

6.4.1.4.2 Reflood Mass and Energy Release Data

The WREFLOOD code is used for computing the reflood transient. The WREFLOOD code consists of two basic hydraulic models—one for the contents of the reactor vessel and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena, such as pumped safety injection and accumulators, reactor coolant pump performance, and steam generator releases are included as auxiliary equations that interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flowrates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break, the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and ECCS injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the usage and application of the Reference 1 M&E release evaluation model in recent analyses, for example, D. C. Cook Docket (Reference 3). Even though the Reference 1 model credits steam/water mixing only in the intact loop and not in the broken loop, the justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 3). Moreover, this assumption is supported by test data and is further discussed below.

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two-phase interaction with condensation of steam by cold ECCS water. The second is a single-phase mixing of condensate and ECCS water. Since the steam release is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that needs to be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data, generated in 1/3-scale tests (Reference 4), are the largest scale data available and thus, most clearly simulate the flow regimes and gravitational effects that would occur in a Pressurized Water Reactor (PWR). These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

A group of 1/3-scale tests corresponds directly to containment integrity reflood conditions. The injection flowrates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 1. For all of these tests, the data clearly indicate the occurrence of very effective mixing with rapid

steam condensation. The mixing model used in the containment integrity reflood calculation is therefore wholly supported by the 1/3-scale steam/water mixing data.

Additionally, the following justification is also noted. The double-ended pump suction break results in the highest containment pressure post-blowdown. For this break, there are two flowpaths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam that is not condensed by ECCS injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECCS injection water as it passes through the broken loop cold leg, complete mixing occurs and a portion of it is condensed. It is this portion of steam that is condensed that is taken credit for in this analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECCS injection nozzle. Descriptions of the test and test results are contained in References 1 and 3.

Table 6.4.1-6 presents the calculated M&E release for the reflood phase of the pump suction double-ended rupture with minimum safeguards. Table 6.4.1-9 presents the calculated M&E release for the reflood phase of the pump suction double-ended rupture with maximum safeguards.

The transient responses of the principal parameters during reflood are given in Table 6.4.1-7 for the DEPS minimum safeguards case. The transient responses of the principal parameters during reflood are given in Table 6.4.1-10 for the DEPS maximum safeguards case.

6.4.1.4.3 Post-Reflood Mass and Energy Release Data

The FROTH code (Reference 5) is used for computing the post-reflood transient. The FROTH code calculates the heat release rates resulting from a two-phase mixture present in the steam generator tubes. The M&E releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken-loop and intact-loop steam generators. During this phase of the transient, the RCS has equilibrated with the containment pressure, but the steam generators contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to core decay heat. For a pump suction break, a two-phase fluid exits the core, flows through the hot legs, and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two-phase. During the FROTH calculation, ECCS injection is addressed for both the injection phase and the recirculation phase. The FROTH code calculation stops when the secondary side equilibrates to the saturation temperature (T_{sat}) at the containment design pressure. After this point, the EPITOME code completes the SG depressurization. (See subsection 6.4.1.4.5 for additional information.)

The methodology for the use of this model is described in Reference 1. The M&E release rates are calculated by FROTH and EPITOME until the time of containment depressurization. After

containment depressurization (14.7 psia), the M&E release available to containment is generated directly from core boil-off/decay heat.

Table 6.4.1-8 presents the two-phase post-reflood M&E release data for the pump suction double-ended case minimum safeguards case. Table 6.4.1-11 presents the two-phase post-reflood M&E release data for the pump suction double-ended maximum safeguards case.

6.4.1.4.4 Decay Heat Model

On November 2, 1978, the Nuclear Power Plant Standards Committee (NUPPSCO) of the American Nuclear Society approved ANS Standard 5.1 (Reference 6) for the determination of decay heat. This standard was used in the M&E release. Table 6.4.1-12 lists the decay heat curve used in the M&E release analysis, post blowdown, for the HNP SGR/Uprating program.

Significant assumptions in the generation of the decay heat curve for use in the LOCA M&E releases analysis include the following:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

2. Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.

3. Fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Equation 11 of Reference 6, up to 10,000 seconds and from Table 10 of Reference 6, beyond 10,000 seconds.

5. The fuel has been assumed to be at full power for 10^8 seconds.

6. The number of atoms of U-239 produced per second has been assumed to be equal to 70 percent of the fission rate.

7. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

8. Two-sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

Based upon NRC staff review, Safety Evaluation Report (SER) of the March 1979 evaluation model (Reference 1), use of the ANS Standard-5.1, November 1979 decay heat model was approved for the calculation of M&E releases to the containment following a LOCA.

6.4.1.4.5 Steam Generator Equilibration and Depressurization

Steam generator equilibration and depressurization is the process by which secondary side energy is removed from the steam generators in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature is the saturation temperature (T_{sat}) at the containment design pressure. After the FROTH calculations, the EPITOME code continues the FROTH calculation for SG cooldown removing steam generator secondary energy at different rates (i.e., first and second stage rates). The first stage rate is applied until the steam generator reaches T_{sat} at the user specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the actual containment pressure. Then the second stage rate is used until the final depressurization, when the secondary reaches the reference temperature of T_{sat} at 14.7 psia, or 212°F. The heat removal of the broken-loop and intact-loop steam generators are calculated separately.

During the FROTH calculations, steam generator heat removal rates are calculated using the secondary side temperature, primary side temperature, and a secondary side heat transfer coefficient determined using a modified McAdam's correlation. Steam generator energy is removed during the FROTH transient until the secondary side temperature reaches saturation temperature at the containment design pressure. The constant heat removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The SG energy available to be released during the first stage interval is determined by calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user specified intermediate equilibration pressure, assuming saturated conditions. The intermediate equilibrium pressures are chosen as discussed in Reference 1, Sections 2.3 and 3.3. This energy is then divided by the first stage energy removal rate, resulting in an intermediate equilibration time. At this time, the rate of energy release drops substantially to the second stage rate. The second stage rate is determined as the fraction of the difference in secondary energy available between the intermediate equilibration and final depressurization at 212°F, and the time difference from the time of the intermediate equilibration to the user-specified time of the final depressurization at 212°F. With current methodology (Reference 1), all of the secondary energy remaining after the intermediate equilibration is conservatively assumed to be released by imposing a mandatory cooldown and subsequent depressurization down to atmospheric pressure at 3600 seconds, i.e., 14.7 psia and 212°F.

6.4.1.4.6 Sources of Mass and Energy

The sources of mass considered in the LOCA M&E release analysis are given in Tables 6.4.1-13, and 6.4.1-14 and 6.4.1-15. These sources are the reactor coolant system, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA M&E release analysis are given in Tables 6.4.1-16, 6.4.1-17 and 6.4.1-18. The energy sources are listed below.

- RCS water
- Accumulator water (all three inject)
- Pumped SI water
- Decay heat
- Core stored energy
- RCS metal (includes SG tubes)
- SG metal (includes transition cone, shell, wrapper, and other internals)
- SG secondary energy (includes fluid mass and steam mass)
- Secondary transfer of energy (feedwater into, and steam out of, the SG secondary)

The energy reference points are as follows.

- Available energy: 212°F; 14.7 psia
- Total energy content: 32°F; 14.7 psia

The mass and energy inventories are presented at the following times, as appropriate:

- Time zero (initial conditions)
- End of blowdown time
- End of refill time
- End of reflood time
- Time of broken loop steam generator equilibration to pressure setpoint
- Time of intact loop steam generator equilibration to pressure setpoint
- Time of full depressurization (3600 seconds)

In the M&E release data presented, no Zirc-water reaction heat was considered because the clad temperature is assumed not to rise high enough for the rate of the Zirc-water reaction heat to be of any significance.

The sequence of events for the LOCA transients are shown in Tables 6.4.1-19 through 6.4.1-21.

6.4.1.5 Conclusions

The consideration of the various energy sources in the long-term mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Thus, the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied. The results of this analysis were provided for use in the containment integrity analysis. (See BOP Licensing Report.) Further, these analyses performed at 2912.4 MWt bound operation at 2787.4 MWt with Westinghouse Δ75 replacement steam generators. Operation at 2787.4 MWt is bounded by the 2912.4 MWt analysis due to the higher core-stored energy and increased level of decay heat power during the long-term transient, including effects on steam generator secondary conditions, which results in higher long-term M&E releases.

Table 6.4.1-1 System Parameters Initial Conditions For Thermal Uprate	
Parameters	**Value**
Core Thermal Power (MWt)	2900
Reactor Coolant System Total Flowrate (lbm/sec)	28833.33
Vessel Outlet Temperature (°F)	623.2
Core Inlet Temperature (°F)	554.4
Vessel Average Temperature (°F)	588.8
Initial Steam Generator Steam Pressure (psia)	1011
Steam Generator Design	Δ75
Steam Generator Tube Plugging (percent)	0
Initial Steam Generator Secondary Side Mass (lbm)	135948
Assumed Maximum Containment Backpressure (psia)	59.7
Accumulator	
Water Volume (ft^3) per accumulator	1029.4
N_2 Cover Gas Pressure (psia)	555
Temperature (°F)	130
Safety Injection Delay, total (sec) (from SI Signal)	29.0

Note:

Core Thermal Power, RCS Total Flowrate, RCS Coolant Temperatures, and Steam Generator Secondary Side Mass include appropriate uncertainty and/or allowance.

Table 6.4.1-2 Safety Injection Flow Minimum Safeguards	
RCS Pressure (psig)	**Total Flow (lbm/sec)**
Injection Mode (Reflood Phase)	
0	629.70
20	586.26
40	538.85
60	485.47
80	423.09
100	344.75
120	230.12
140	70.57
160	70.28
180	69.99
Injection Mode (Post-Reflood Phase)	
45	525.50
Cold Leg Recirculation Mode	
45	470.00

Table 6.4.1-3 Safety Injection Flow Maximum Safeguards	
RCS Pressure (psig)	**Total Flow (lbm/sec)**
Injection Mode (Reflood Phase)	
0	997.50
45	860.85
100	646.48
110	597.37
120	534.19
Injection Mode (Post-Reflood Phase)	
45	860.85
Cold Leg Recirculation Mode	
45	~800

Table 6.4.1-4 Double-Ended Hot-Leg Break Blowdown M&E Releases				
Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Btu/sec)	(lbm/sec)	(Btu/sec)
0.00000	0.00	0.0	0.00	0.00
0.00104	45868.80	30054200.0	45867.69	30052600.00
0.00216	47552.30	31154100.0	47199.60	30916500.00
0.102	39304.00	26044600.0	28042.90	18333800.00
0.201	37478.50	24779600.0	24280.90	15764100.00
0.302	36093.10	23828900.0	21852.30	13996700.00
0.401	34827.60	22987600.0	20649.40	13017000.00
0.501	34050.39	22462800.0	19856.90	12327000.00
0.602	33959.89	22398000.0	19377.19	11866300.00
0.701	33597.50	22187200.0	19006.90	11507400.00
0.801	32894.80	21780400.0	18720.30	11224700.00
0.902	32500.00	21606200.0	18435.90	10965400.00
1.00	32065.09	21429400.0	18227.50	10767100.00
1.10	31559.40	21210000.0	18062.50	10606700.00
1.20	30994.00	20947500.0	17948.50	10484300.00
1.30	30430.30	20688500.0	17876.30	10392500.00
1.40	29856.00	20423000.0	17849.80	10333100.00
1.50	29225.80	20121400.0	17855.80	10296200.00
1.60	28505.90	19756300.0	17884.59	10275700.00
1.70	27723.59	19341000.0	17922.30	10263800.00
1.80	26936.80	18918500.0	17964.00	10257100.00
1.90	26202.19	18529400.0	18006.09	10253600.00
2.00	25494.90	18153900.0	18044.00	10251000.00
2.10	24774.19	17754400.0	18073.30	10246500.00
2.20	24072.50	17354000.0	18092.30	10239400.00
2.30	23379.90	16945900.0	18098.09	10227400.00
2.40	22718.19	16545199.0	18087.30	10209000.00
2.50	22125.90	16177700.0	18060.50	10184000.00
2.60	21586.00	15827900.0	18014.40	10150300.00
2.70	21127.09	15516800.0	17950.50	10108700.00
2.80	20729.09	15230200.0	17868.90	10059000.00
2.90	20386.30	14967800.0	17768.09	10000300.00
3.00	20100.30	14732100.0	17649.19	9932900.00
3.10	19873.50	14528200.0	17515.50	9858500.00
3.20	19692.69	14347400.0	17363.59	9775100.00
3.30	19557.00	14191500.0	17195.00	9683400.00
3.40	19460.00	14058900.0	17007.69	9582000.00
3.50	19396.19	13946000.0	16797.69	9468900.00

Table 6.4.1-4 (Cont.) Double-Ended Hot-Leg Break Blowdown M&E Releases				
Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Btu/sec)	(lbm/sec)	(Btu/sec)
3.60	19365.09	13853000.0	16580.09	9352300.000
3.70	19371.19	13783400.0	16360.50	9235200.000
3.80	19406.19	13730100.0	16137.20	9116700.000
3.90	19464.90	13696300.0	15917.00	9000200.000
4.00	19532.90	13672300.0	15685.70	8877900.000
4.20	19738.80	13650900.0	15265.70	8657900.000
4.40	19993.50	13655000.0	14869.70	8451300.000
4.60	20282.09	13692500.0	14314.59	8152400.000
4.80	20610.50	13752700.0	13779.70	7867500.000
5.00	21057.19	13864400.0	13258.29	7592000.000
5.20	12936.50	9460400.0	12775.70	7339500.000
5.40	16074.40	11456900.0	12331.70	7108000.000
5.60	16312.29	11507500.0	11897.00	6880800.000
5.80	16516.59	11535400.0	11527.59	6690300.000
6.00	16730.59	11581900.0	11189.40	6515200.000
6.20	16961.30	11578800.0	10850.40	6337000.000
6.40	17183.00	11604200.0	10548.59	6179000.000
6.60	17299.30	11595600.0	10266.90	6030900.000
6.80	17203.30	11489200.0	10006.40	5893700.000
7.00	17385.30	11508300.0	9758.200	5762500.000
7.20	17605.09	11553900.0	9508.900	5629400.000
7.40	17806.00	11594700.0	9262.299	5497600.000
7.60	18010.40	11641500.0	9017.700	5367100.000
7.80	18240.90	11706900.0	8769.200	5234700.000
8.00	18410.80	11752900.0	8525.200	5105600.000
8.20	18093.80	11523000.0	8276.299	4974500.000
8.40	16382.50	10567800.0	8027.399	4844200.000
8.60	15441.70	10032000.0	7782.500	4717300.000
8.80	15336.09	9939700.0	7540.500	4593400.000
9.00	15305.79	9897600.0	7311.000	4477700.000
9.20	15255.20	9849800.0	7090.299	4367300.000
9.40	15139.79	9768900.0	6878.500	4261700.000
9.60	14873.79	9607300.0	6671.399	4158299.750
9.80	14325.79	9294700.0	6468.000	4057400.000
10.0	13642.59	8911300.0	6269.299	3959500.000
10.2	13160.20	8636500.0	6075.000	3864600.000
10.4	12836.50	8449900.0	5886.799	3774000.000
10.6	12542.00	8283700.0	5703.799	3686800.000

Table 6.4.1-4 (Cont.) Double-Ended Hot-Leg Break Blowdown M&E Releases				
Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Btu/sec)	(lbm/sec)	(Btu/sec)
10.8	12215.90	8105200.00	5526.600	3603000.000
11.0	11846.09	7906800.00	5356.399	3522700.000
11.2	11438.29	7691400.00	5191.000	3444800.000
11.4	11026.79	7477100.00	5032.200	3370000.000
11.6	10630.70	7275200.00	4878.700	3298400.000
11.8	10133.59	7091500.00	4730.100	3229600.000
12.0	9339.200	6857300.00	4584.299	3162400.000
12.2	8865.500	6706800.00	4443.899	3097700.000
12.4	8414.700	6523200.00	4304.299	3033200.000
12.6	7830.600	6240600.00	4164.399	2968600.000
12.8	7226.100	5943700.00	4025.199	2905300.000
13.0	6764.700	5739800.00	3885.899	2844000.000
13.2	6322.100	5415900.00	3741.600	2781600.000
13.4	5889.100	5080400.00	3581.399	2711600.000
13.6	5413.200	4793700.00	3396.800	2631900.000
13.8	4893.700	4497700.00	3174.100	2543800.000
14.0	4385.600	4205400.00	2914.800	2453500.000
14.2	3924.500	3913700.00	2625.300	2363600.000
14.4	3520.199	3660600.00	2318.399	2276700.000
14.6	3188.000	3443000.00	2016.599	2180300.000
14.8	2931.500	3256300.00	1759.300	2059700.000
15.0	2724.399	3074500.00	1562.500	1901400.000
15.4	2277.699	2665900.00	1300.699	1611600.000
15.6	2050.399	2439800.00	1200.599	1492700.000
15.8	1842.099	2218300.00	1116.099	1391100.000
16.0	1635.900	1986300.00	1033.900	1291300.000
16.2	1459.800	1787800.00	950.4000	1189200.000
16.4	1319.500	1631300.00	882.4000	1106500.000
16.6	1240.099	1547600.00	824.2000	1035600.000
16.8	1164.699	1462300.00	771.7000	971000.000
17.0	1072.099	1349000.00	728.0000	917300.000
17.4	911.9000	1149900.00	658.9000	832000.000
17.6	842.0000	1063000.00	633.0000	800100.000
17.8	745.5000	942400.00	614.2999	777200.000
18.0	655.7999	831100.00	603.2999	763800.000
18.2	577.4000	732400.00	596.0999	755200.000
18.4	497.6000	632700.00	589.0000	746500.000
18.6	414.5000	527600.000	571.9000	724900.0000
18.8	333.7999	425500.000	530.0000	672000.0000

Table 6.4.1-4 (Cont.) Double-Ended Hot-Leg Break Blowdown M&E Releases				
Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Btu/sec)	(lbm/sec)	(Btu/sec)
19.0	231.8999	296100.000	476.3999	604600.0000
19.2	125.8000	161000.000	441.2999	561000.0000
19.4	0.0000	0.000	421.3999	536200.0000
19.6	0.0000	0.000	344.6000	438300.0000
19.8	0.0000	0.000	248.3999	316800.0000
20.0	0.0000	0.000	146.3999	187300.0000
20.2	0.0000	0.000	0.0000	0.0000

* M&E exiting from the RV side of the break

** M&E exiting from the SG side of the break

Table 6.4.1-5
Double-Ended Pump Suction Break
Blowdown M&E Releases
(Same for all DEPS Runs)

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
0.00000	0.0	0.0	0.0	0.0
0.00102	92463.8	51485.9	40808.9	22659.2
0.00209	42572.8	23639.1	42166.4	23411.3
0.101	42230.4	23509.3	21593.0	11977.5
0.201	43036.6	24088.9	24339.8	13513.6
0.301	45457.6	25620.3	24771.9	13767.7
0.501	45811.1	26285.6	22806.4	12697.9
0.701	45947.9	26903.5	21102.6	11756.3
0.902	45086.2	26887.8	20243.6	11286.3
1.20	41912.3	25628.0	19759.7	11019.8
2.00	34787.9	22749.2	19342.1	10781.9
2.30	30697.4	20945.5	18738.3	10443.4
2.40	28263.4	19597.2	18335.2	10217.0
2.50	24063.3	16949.4	17767.9	9900.3
2.60	21095.3	15133.5	17472.9	9737.4
2.80	17157.2	12691.7	16980.6	9466.0
3.00	15029.1	11341.2	16485.0	9194.2
3.20	13842.7	10580.6	16106.8	8989.3
3.50	12676.2	9845.0	15550.9	8688.6
4.00	11360.4	9072.2	14676.1	8218.1
4.40	10583.5	8612.8	14578.8	8184.8
4.60	10270.6	8412.5	15940.2	8956.5
5.20	9807.0	8030.8	15032.2	8473.6
5.60	9623.0	7799.4	14607.6	8252.7
6.00	10323.8	8371.1	14478.0	8182.5
6.40	8675.0	7822.4	13902.4	7842.3
6.60	8565.5	7627.3	13748.1	7754.5
7.00	8889.9	7434.6	13352.2	7524.9
7.80	9919.5	7427.4	12618.0	7095.4
8.40	9731.5	7070.2	12176.3	6837.1
9.80	8255.6	6168.3	11248.9	6300.8
11.0	6936.7	5332.5	10207.0	5715.0
13.2	5686.6	4439.0	8872.5	5013.5
14.0	5240.2	4397.4	7716.0	4524.6
14.6	4334.0	4340.9	6611.3	3819.7
15.0	3377.8	3948.0	5996.9	3142.1
15.4	2633.2	3253.1	5328.1	2578.0
15.8	2068.6	2580.7	4590.7	2104.4
16.2	1673.1	2099.8	3986.9	1736.5

Table 6.4.1-5 (Cont.) Double-Ended Pump Suction Break Blowdown M&E Releases (Same for all DEPS Runs)				
Time (se)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
16.6	1426.1	1797.1	2580.7	1019.1
16.8	1314.9	1659.6	2272.5	836.3
17.4	967.6	1226.2	2389.1	784.6
17.8	719.0	913.8	3202.4	998.9
18.0	521.7	663.3	3219.9	977.8
18.6	0.0	0.0	1313.0	388.5
19.0	0.0	0.0	234.3	69.8
19.6	0.0	0.0	398.6	121.3
20.0	0.0	0.0	253.5	78.2
21.0	0.0	0.0	237.8	75.3
21.4	0.0	0.0	0.0	0.0

* M&E exiting the SG side of the break

** M&E exiting the pump side of the break

Table 6.4.1-6
Double-Ended Pump Suction Break
Minimum Safeguards
Reflood M&E Releases

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
21.4	0.0	0.0	0.0	0.0
22.3	0.0	0.0	0.0	0.0
22.5	43.9	51.7	0.0	0.0
22.6	28.1	33.1	0.0	0.0
22.8	35.7	42.1	0.0	0.0
23.3	69.3	81.7	0.0	0.0
25.5	136.7	161.1	0.0	0.0
26.0	147.5	173.9	0.0	0.0
26.5	380.3	450.0	3827.1	514.9
27.6	435.3	515.6	4380.7	609.0
28.6	425.8	504.3	4283.7	599.5
30.6	404.7	479.1	4069.3	576.2
31.6	394.5	467.0	3964.7	564.5
32.7	413.8	489.9	4188.8	583.3
33.7	402.8	476.9	4071.0	572.1
34.7	394.2	466.6	3982.5	562.0
35.7	386.0	456.9	3897.4	552.2
37.7	370.7	438.6	3736.8	533.8
39.7	356.8	422.0	3587.9	516.7
41.7	344.0	406.8	3449.2	500.7
42.7	338.0	399.6	3383.4	493.1
44.7	326.6	386.1	3257.8	478.5
46.7	316.1	373.6	3139.6	464.8
48.7	306.3	361.9	3028.1	451.8
50.7	297.1	351.0	2922.6	439.5
52.7	288.5	340.8	2822.4	427.8
53.7	232.7	274.7	2108.8	351.7
54.8	303.4	358.5	277.4	148.5
60.8	278.4	328.9	268.3	135.9
61.8	274.3	324.0	266.8	133.9
65.8	259.7	306.7	261.6	126.7
69.8	246.2	290.7	256.8	120.2
73.8	233.8	275.9	252.5	114.3
81.8	212.0	250.1	245.1	104.2
82.8	209.5	247.2	244.3	103.1
90.8	192.2	226.7	238.6	95.4
98.8	178.6	210.6	234.2	89.5
108.8	166.1	195.8	230.2	84.2
112.8	162.3	191.3	229.0	82.7

Table 6.4.1-6 (Cont.) Double-Ended Pump Suction Break Minimum Safeguards Reflood M&E Releases				
Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
124.8	154.2	181.8	226.5	79.3
138.8	149.6	176.3	225.0	77.3
152.8	148.1	174.6	224.4	76.6
164.8	148.2	174.7	224.3	76.4
180.8	149.5	176.3	224.6	76.7
184.8	150.9	177.9	225.8	77.3
192.8	153.3	180.7	233.1	79.4
200.8	154.8	182.5	243.6	81.8
206.8	154.9	182.6	252.3	83.4
214.8	153.5	180.9	264.8	85.4
216.6	152.9	180.3	267.8	85.8

* M&E exiting the SG side of the break

** M&E exiting the pump side of the break

Table 6.4.1-7
Double-Ended Pump Suction Break, Minimum Safeguards
Principal Parameters During Reflood

Time (sec)	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (lbm/sec)	Injection Accum (lbm/sec)	Spill (lbm/sec)	Enthalpy (Btu/lbm)
21.4	223.6	0.000	0.000	0.00	0.00	0.333	0.0	0.0	0.0	0.00
22.2	219.0	26.896	0.000	0.80	1.83	0.000	7558.9	7558.9	0.0	99.35
22.3	217.9	28.689	0.000	1.03	1.82	0.000	7521.3	7521.3	0.0	99.35
23.5	216.0	2.751	0.316	1.50	5.91	0.430	7041.5	7041.5	0.0	99.35
24.4	215.7	2.651	0.436	1.63	9.10	0.451	6771.0	6771.0	0.0	99.35
27.6	214.4	4.679	0.654	2.03	15.59	0.684	5293.1	5293.1	0.0	99.35
28.6	214.0	4.409	0.682	2.15	15.59	0.682	5115.4	5115.4	0.0	99.35
31.6	213.0	3.928	0.722	2.45	15.59	0.671	4675.1	4675.1	0.0	99.35
32.7	212.8	4.016	0.731	2.55	15.59	0.683	4915.3	4453.5	0.0	98.75
38.7	212.4	3.544	0.750	3.03	15.59	0.661	4280.7	3804.4	0.0	98.64
45.7	213.1	3.197	0.757	3.51	15.59	0.642	3742.6	3253.6	0.0	98.52
53.7	214.9	2.607	0.756	4.01	15.59	0.571	2519.4	2008.9	0.0	98.06
54.8	215.1	3.063	0.761	4.07	15.44	0.630	493.5	0.0	0.0	92.99
62.8	218.3	2.766	0.761	4.53	14.37	0.622	500.1	0.0	0.0	92.99
71.8	223.4	2.493	0.760	5.00	13.50	0.613	505.7	0.0	0.0	92.99
82.8	230.8	2.225	0.759	5.52	12.85	0.601	510.8	0.0	0.0	92.99
94.8	239.0	2.009	0.759	6.03	12.53	0.587	514.5	0.0	0.0	92.99
108.8	246.7	1.842	0.759	6.57	12.51	0.575	517.1	0.0	0.0	92.99
120.8	252.3	1.755	0.761	7.00	12.69	0.567	518.3	0.0	0.0	92.99
136.8	258.6	1.691	0.765	7.55	13.10	0.562	519.2	0.0	0.0	92.99
150.8	263.4	1.666	0.769	8.00	13.54	0.560	519.5	0.0	0.0	92.99
166.8	268.1	1.654	0.774	8.51	14.09	0.561	519.5	0.0	0.0	92.99
176.8	270.8	1.653	0.778	8.82	14.44	0.562	519.5	0.0	0.0	92.99
182.8	272.3	1.657	0.780	9.00	14.65	0.563	519.4	0.0	0.0	92.99
196.8	275.5	1.671	0.785	9.42	15.08	0.569	518.8	0.0	0.0	92.99
200.8	276.3	1.670	0.787	9.54	15.17	0.571	518.8	0.0	0.0	92.99
216.6	279.4	1.632	0.791	10.00	15.43	0.573	518.9	0.0	0.0	92.99

Table 6.4.1-8 Double-Ended Pump Suction Break Minimum Safeguards, Post-Reflood M&E Releases				
	Break Path No. 1 Flow		Break Path No. 2 Flow	
Time (sec)	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
216.6	148.9	188.1	377.0	107.0
221.6	148.5	187.6	377.4	107.0
301.6	141.5	178.8	384.4	106.9
421.6	130.0	164.3	395.9	106.8
456.6	127.8	161.4	398.1	106.5
496.6	122.5	154.8	403.4	106.9
521.6	120.9	152.8	404.9	106.6
561.6	117.2	148.0	408.7	106.6
566.6	118.1	149.3	407.8	106.2
581.6	117.1	148.0	408.7	106.1
586.6	118.1	149.2	407.8	105.7
606.6	116.8	147.6	409.1	105.5
611.6	117.7	148.8	408.2	105.1
631.6	116.5	147.2	409.4	104.9
636.6	117.4	148.3	408.5	104.5
651.6	116.4	147.1	409.5	104.4
656.6	117.3	148.2	408.6	104.0
671.6	116.3	146.9	409.6	103.9
676.6	117.1	148.0	408.8	103.5
691.6	116.0	146.6	409.8	103.4
711.6	116.8	147.6	409.1	102.6
721.6	116.0	146.6	409.8	102.6
741.6	116.6	147.3	409.3	101.9
766.6	115.5	145.9	410.4	101.5
781.6	116.1	146.7	409.8	100.9
821.6	115.0	145.3	410.9	104.1
831.6	115.7	146.2	410.2	103.6
846.6	114.8	145.0	411.1	103.4
861.6	115.4	145.8	410.5	102.7
946.6	114.3	144.4	411.6	100.2
951.6	65.4	82.6	460.5	112.9
1247.0	65.4	82.6	460.5	112.9
1247.1	71.3	88.9	454.6	110.8
1805.2	71.2	88.9	454.7	110.6
1805.3	64.7	74.5	461.2	52.0
2210.0	61.8	71.1	464.1	52.6
2210.1	61.8	71.1	437.4	78.1
3600.0	54.7	63.0	444.5	79.4
3600.1	43.4	49.9	455.8	68.0
10000.0	31.2	35.9	469.8	65.3
18000.0	26.4	30.4	476.8	60.1
18001.1	26.5	30.5	475.9	62.4

Table 6.4.1-8 (Cont.) Double-Ended Pump Suction Break Minimum Safeguards, Post-Reflood M&E Releases				
Time (sec)	Break Path No. 1 Flow		Break Path No. 2 Flow	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
30000.0	23.4	26.9	480.2	59.6
86400.0	17.1	19.6	487.8	56.6
100000.0	16.3	18.7	488.8	55.7
106400.0	16.0	18.4	489.3	55.3
1000000.0	6.9	7.9	500.5	49.6
2592000.0	4.65	5.35	502.7	49.8
10000000.0	2.2	2.5	505.2	50.06
32000000.0	1.7	1.96	505.6	50.1

Table 6.4.1-9
Double-Ended Pump Suction Break
Maximum Safeguards
Reflood M&E Releases

Time (sec)	Break Path No. 1 Flow*		Break Path No. 2 Flow**	
	(lbm/sec)	(Thousand Btu/sec)	(lbm/sec)	(Thousand Btu/sec)
21.4	.0	.0	.0	.0
21.9	.0	.0	.0	.0
22.0	.0	.0	.0	.0
22.2	.0	.0	.0	.0
22.3	.0	.0	.0	.0
22.5	43.9	51.7	.0	.0
22.5	32.2	37.9	.0	.0
22.6	28.1	33.1	.0	.0
22.7	30.9	36.4	.0	.0
22.8	35.7	42.1	.0	.0
22.9	43.1	50.8	.0	.0
23.0	49.8	58.7	.0	.0
23.1	56.7	66.7	.0	.0
23.2	62.8	74.0	.0	.0
23.3	69.3	81.7	.0	.0
23.4	73.3	86.4	.0	.0
23.5	77.3	91.1	.0	.0
23.6	81.2	95.6	.0	.0
23.7	84.8	99.9	.0	.0
23.8	88.4	104.1	.0	.0
23.9	91.9	108.2	.0	.0
24.0	95.2	112.2	.0	.0
24.1	98.5	116.0	.0	.0
24.2	101.6	119.7	.0	.0
24.3	104.7	123.4	.0	.0
24.4	107.7	126.9	.0	.0
24.5	110.6	130.3	.0	.0
25.5	136.7	161.1	.0	.0
26.0	147.5	173.9	.0	.0
26.5	380.3	450.0	3827.1	514.9
27.6	435.3	515.6	4380.7	609.0
28.6	425.8	504.3	4283.7	599.5
29.6	415.2	491.6	4176.5	588.0
30.0	410.9	486.6	4133.4	583.3
30.6	404.7	479.1	4069.3	576.2
31.6	394.5	467.0	3964.7	564.5
32.7	434.5	514.6	4406.8	604.6
33.7	421.1	498.7	4269.0	591.6
34.7	412.6	488.5	4182.8	581.8

Table 6.4.1-9 (Cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
35.6	405.2	479.8	4108.1	573.2
35.7	404.4	478.8	4099.9	572.2
36.7	396.7	469.5	4020.0	563.1
37.7	389.2	460.7	3943.2	554.2
38.7	382.1	452.2	3869.1	545.6
39.7	375.3	444.1	3797.6	537.4
40.7	368.8	436.4	3728.7	529.4
41.7	362.6	428.9	3662.1	521.7
42.2	359.5	425.3	3629.6	517.9
42.7	356.6	421.8	3597.7	514.2
43.7	350.8	414.9	3535.4	507.0
44.7	345.2	408.3	3475.0	500.0
45.7	339.9	401.9	3416.5	493.2
46.7	334.7	395.7	3359.7	486.6
47.7	329.7	389.8	3304.5	480.1
48.7	324.9	384.1	3250.9	473.9
49.7	320.2	378.5	3198.7	467.8
50.7	315.7	373.1	3147.9	461.8
51.7	311.3	367.9	3098.4	456.0
52.7	307.1	362.9	3050.2	450.4
53.7	303.0	358.0	3003.1	444.8
54.7	205.5	242.4	500.9	141.9
55.7	208.9	246.5	494.1	141.2
56.7	207.5	244.8	496.7	141.4
57.7	206.0	243.1	499.3	141.7
58.6	204.7	241.5	501.7	141.9
58.7	204.5	241.3	502.0	142.0
59.7	203.0	239.5	504.7	142.3
60.7	201.5	237.7	507.4	142.6
61.7	199.9	235.8	510.2	142.9
62.7	198.3	233.9	513.0	143.2
63.7	196.7	232.0	515.8	143.5
64.7	195.1	230.1	518.6	143.8
65.7	193.4	228.1	521.5	144.2
66.7	191.7	226.1	524.5	144.5
67.7	190.0	224.1	527.4	144.9
68.7	188.3	222.0	530.5	145.2
69.7	186.5	219.9	533.5	145.6

Table 6.4.1-9 (Cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
70.7	184.6	217.7	536.6	146.0
71.7	182.8	215.6	539.8	146.4
72.7	180.9	213.3	543.0	146.8
73.7	179.0	211.0	546.3	147.2
74.7	177.0	208.7	549.6	147.7
75.7	175.0	206.3	553.0	148.1
76.7	172.9	203.9	556.4	148.6
77.7	170.8	201.4	559.9	149.1
78.7	168.7	198.9	563.5	149.6
79.7	166.5	196.3	567.1	150.1
80.4	164.9	194.4	569.7	150.5
80.7	164.2	193.6	570.8	150.6
81.7	161.9	190.9	574.7	151.2
82.7	159.5	188.0	578.6	151.8
84.7	157.8	186.0	581.8	151.8
86.7	157.2	185.3	583.2	151.6
88.7	156.6	184.6	584.7	151.3
90.7	156.0	183.9	586.1	151.1
92.7	155.5	183.2	587.5	150.9
94.7	154.9	182.6	588.9	150.6
96.7	154.3	181.9	590.4	150.4
98.7	153.7	181.2	591.8	150.1
100.7	153.1	180.5	593.2	149.9
102.7	152.5	179.8	594.6	149.6
104.7	151.9	179.1	596.0	149.4
105.4	151.7	178.9	596.4	149.3
106.7	151.3	178.4	597.3	149.1
108.7	150.8	177.7	598.7	148.9
110.7	150.2	177.0	600.1	148.6
112.7	149.6	176.3	601.5	148.4
114.7	148.9	175.6	602.9	148.1
114.7	148.9	175.6	602.9	148.1
116.7	148.3	174.8	604.3	147.8
118.7	147.7	174.1	605.6	147.6
120.7	147.1	173.4	607.0	147.3
122.7	146.5	172.7	608.4	147.1
124.7	145.9	172.0	609.7	146.8
126.7	145.3	171.2	611.1	146.5
128.7	144.7	170.5	612.4	146.3
130.7	144.1	169.8	613.8	146.0

Table 6.4.1-9 (Cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
132.7	143.4	169.1	615.1	145.8
132.8	143.4	169.0	615.2	145.7
134.7	142.8	168.3	616.5	145.5
136.7	142.2	167.6	617.8	145.2
138.7	141.6	166.9	619.2	145.0
140.7	141.0	166.1	620.5	144.7
142.7	140.3	165.4	621.9	144.4
144.7	139.7	164.7	623.2	144.2
146.7	139.1	163.9	624.5	143.9
148.7	138.5	163.2	625.9	143.6
150.7	137.8	162.4	627.2	143.4
152.7	137.2	161.7	628.5	143.1
154.7	136.5	160.9	629.9	142.8
156.7	135.9	160.2	631.2	142.6
158.7	135.3	159.4	632.5	142.3
160.7	134.6	158.6	633.9	142.0
162.7	134.0	157.9	635.2	141.8
163.0	133.9	157.8	635.4	141.7
164.7	133.3	157.1	636.5	141.5
166.7	132.7	156.4	637.8	141.2
168.7	132.0	155.6	639.2	141.0
170.7	131.4	154.8	640.5	140.7
172.7	130.7	154.1	641.8	140.4
174.7	130.1	153.3	643.1	140.2
176.7	129.4	152.5	644.4	139.9
178.7	128.8	151.8	645.7	139.7
180.7	128.1	151.0	647.1	139.4
182.7	127.5	150.2	648.4	139.1
184.7	126.8	149.4	649.7	138.9
186.7	126.2	148.7	651.0	138.6
188.7	125.5	147.9	652.3	138.3
190.7	124.9	147.1	653.6	138.1
192.7	124.2	146.4	654.9	137.8
194.7	123.6	145.6	656.2	137.6
196.7	122.9	144.8	657.5	137.3
197.0	122.8	144.7	657.7	137.3
198.7	122.2	144.0	658.8	137.1
200.7	121.6	143.3	660.1	136.8
202.7	120.9	142.5	661.4	136.5
204.7	120.3	141.7	662.7	136.3

Table 6.4.1-9 (Cont.) Double-Ended Pump Suction Break Maximum Safeguards Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
206.7	119.6	140.9	664.0	136.0
208.7	118.9	140.1	665.3	135.8
210.7	118.3	139.3	666.6	135.5
212.7	117.6	138.5	667.9	135.3
214.7	116.9	137.8	669.2	135.0
216.7	116.2	137.0	670.5	134.8
218.7	115.6	136.2	671.7	134.5
220.7	114.9	135.4	673.0	134.3
222.7	114.2	134.6	674.3	134.0
223.8	113.9	134.2	675.0	133.9

* M&E exiting the SG side of the break

** M&E exiting the pump side of the break

Table 6.4.1-10
Double-Ended Pump Suction Break, Maximum Safeguards
Principal Parameters During Reflood

Time (sec)	Flooding Temp (°F)	Flooding Rate (in/sec)	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Total (lbm/sec)	Injection Accum (lbm/sec)	Spill (lbm/sec)	Enthalpy (Btu/lbm)
21.4	223.6	.000	.000	.00	.00	.333	.0	.0	.0	.00
22.2	219.0	26.896	.000	.80	1.83	.000	7558.9	7558.9	.0	99.35
22.3	217.9	28.689	.000	1.03	1.82	.000	7521.3	7521.3	.0	99.35
23.5	216.0	2.751	.316	1.50	5.91	.430	7041.5	7041.5	.0	99.35
24.4	215.7	2.651	.436	1.63	9.10	.451	6771.0	6771.0	.0	99.35
27.6	214.4	4.679	.654	2.03	15.59	.684	5293.1	5293.1	.0	99.35
28.6	214.0	4.409	.682	2.15	15.59	.682	5115.4	5115.4	.0	99.35
31.6	213.0	3.928	.722	2.45	15.59	.671	4675.1	4675.1	.0	99.35
32.7	212.8	4.153	.732	2.55	15.59	.692	5163.7	4396.8	.0	98.40
38.7	212.3	3.671	.751	3.05	15.59	.670	4515.8	3732.7	.0	98.25
45.7	212.9	3.323	.758	3.55	15.59	.653	3988.1	3187.7	.0	98.07
53.7	214.6	3.037	.760	4.05	15.59	.635	3514.6	2700.3	.0	97.88
54.7	214.8	2.427	.754	4.11	15.59	.547	848.2	.0	.0	92.99
55.7	215.1	2.439	.754	4.16	15.59	.550	846.4	.0	.0	92.99
62.7	217.8	2.348	.755	4.50	15.59	.541	849.0	.0	.0	92.99
73.7	223.5	2.190	.756	5.01	15.59	.523	853.6	.0	.0	92.99
86.7	231.6	2.014	.756	5.56	15.59	.500	858.3	.0	.0	92.99
98.7	239.2	1.957	.760	6.04	15.59	.501	858.3	.0	.0	92.99
110.7	245.9	1.901	.763	6.50	15.59	.501	858.3	.0	.0	92.99
124.7	252.6	1.836	.766	7.02	15.59	.502	858.3	.0	.0	92.99
138.7	258.2	1.772	.769	7.51	15.59	.503	858.3	.0	.0	92.99
154.7	263.8	1.699	.772	8.04	15.59	.504	858.4	.0	.0	92.99
170.7	268.5	1.626	.776	8.54	15.59	.504	858.5	.0	.0	92.99
186.7	272.4	1.554	.779	9.01	15.59	.505	858.6	.0	.0	92.99
204.7	276.3	1.474	.782	9.51	15.59	.505	858.7	.0	.0	92.99
223.8	279.7	1.389	.786	10.00	15.59	.505	858.9	.0	.0	92.99

Table 6.4.1-11 Double-Ended Pump Suction Break Maximum Safeguards, Post-Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
223.8	125.6	159.2	735.9	142.8
228.8	125.2	158.8	736.2	142.8
233.8	124.9	158.4	736.6	142.8
238.8	126.2	160.0	735.2	142.3
243.8	125.9	159.6	735.6	142.3
248.8	125.5	159.2	735.9	142.2
253.8	125.2	158.7	736.3	142.2
258.8	124.9	158.3	736.6	142.2
263.8	124.5	157.9	737.0	142.1
268.8	124.2	157.4	737.3	142.1
273.8	125.5	159.1	736.0	141.6
278.8	125.1	158.6	736.3	141.6
283.8	124.8	158.2	736.7	141.6
288.8	124.4	157.8	737.0	141.5
293.8	124.1	157.3	737.4	141.5
298.8	123.7	156.9	737.7	141.5
303.8	125.0	158.5	736.5	141.0
308.8	124.6	158.0	736.8	141.0
313.8	124.3	157.6	737.2	140.9
318.8	123.9	157.1	737.5	140.9
323.8	123.6	156.7	737.9	140.9
328.8	124.8	158.3	736.6	140.4
333.8	124.5	157.8	737.0	140.4
338.8	124.1	157.4	737.4	140.4
343.8	123.7	156.9	737.7	140.3
348.8	123.4	156.4	738.1	140.3
353.8	123.0	156.0	738.4	140.3
358.8	124.2	157.5	737.2	139.8
363.8	123.9	157.1	737.6	139.8
368.8	123.5	156.6	738.0	139.8
373.8	123.1	156.1	738.3	139.7
378.8	122.8	155.7	738.7	139.7
383.8	122.4	155.2	739.1	139.7
388.8	123.6	156.7	737.9	139.2
393.8	123.2	156.2	738.2	139.2
398.8	122.8	155.8	738.6	139.2
403.8	122.6	155.4	738.9	139.1
408.8	122.3	155.1	739.2	139.1
413.8	122.1	154.8	739.4	139.0
418.8	123.4	156.4	738.1	138.5

Table 6.4.1-11 (Cont.) Double-Ended Pump Suction Break Maximum Safeguards, Post-Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
423.8	123.1	156.1	738.4	138.5
428.8	122.8	155.7	738.6	138.4
433.8	122.6	155.4	738.9	138.4
438.8	122.3	155.1	739.2	138.3
443.8	122.0	154.7	739.4	138.3
448.8	121.8	154.4	739.7	138.2
453.8	123.1	156.0	738.4	137.8
458.8	122.8	155.7	738.7	137.7
463.8	122.5	155.3	738.9	137.7
468.8	122.2	155.0	739.2	137.6
473.8	122.0	154.6	739.5	137.6
478.8	121.7	154.3	739.8	137.5
483.8	121.4	154.0	740.0	137.4
488.8	122.7	155.5	738.8	137.0
493.8	122.4	155.2	739.1	136.9
498.8	122.1	154.8	739.4	136.9
503.8	121.8	154.4	739.6	136.8
508.8	121.5	154.1	739.9	136.8
513.8	121.2	153.7	740.2	136.7
518.8	122.4	155.3	739.0	136.3
523.8	122.2	154.9	739.3	136.2
528.8	121.9	154.5	739.6	136.2
533.8	121.6	154.1	739.9	136.1
538.8	121.3	153.8	740.2	136.1
543.8	121.0	153.4	740.5	136.0
548.8	122.1	154.9	739.3	135.6
553.8	121.8	154.5	739.6	135.6
558.8	121.5	154.1	739.9	135.5
563.8	121.2	153.7	740.2	135.5
568.8	120.9	153.3	740.5	135.4
573.8	120.6	153.0	740.8	135.4
578.8	121.8	154.4	739.7	134.9
583.8	121.4	154.0	740.0	134.9
588.8	121.1	153.6	740.3	134.8
593.8	120.8	153.2	740.7	134.8
598.8	120.5	152.8	741.0	134.7
603.8	121.6	154.2	739.8	134.3
608.8	121.3	153.8	740.1	134.3
613.8	121.0	153.5	740.4	134.2
618.8	120.8	153.1	740.7	134.2

Table 6.4.1-11 (Cont.) Double-Ended Pump Suction Break Maximum Safeguards, Post-Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
828.8	119.6	151.6	741.9	132.6
833.8	119.1	151.0	742.4	132.6
838.8	118.6	150.4	742.8	132.5
843.8	119.3	151.2	742.2	132.2
848.8	118.8	150.6	742.7	132.2
853.8	119.4	151.4	742.1	131.8
858.8	118.9	150.7	742.6	131.8
863.8	119.4	151.4	742.0	131.5
868.8	118.9	150.8	742.6	131.5
873.8	119.4	151.4	742.0	131.2
878.8	118.9	150.7	742.6	131.2
883.8	119.3	151.3	742.1	130.9
888.8	118.7	150.6	742.7	130.9
893.8	119.2	151.1	742.3	130.6
898.8	118.5	150.3	742.9	130.6
903.8	118.9	150.8	742.6	130.4
908.8	118.3	149.9	743.2	130.4
913.8	118.6	150.3	742.9	130.2
918.8	118.8	150.7	742.6	133.8
923.8	119.1	151.0	742.4	133.6
928.8	118.3	150.0	743.1	133.6
933.8	118.5	150.2	743.0	133.3
938.8	118.6	150.4	742.8	133.1
943.8	118.7	150.5	742.8	132.9
948.8	118.7	150.5	742.8	132.7
953.8	118.7	150.5	742.8	132.6
958.8	118.6	150.4	742.9	132.4
963.8	118.5	150.2	743.0	132.3
968.8	118.3	149.9	743.2	132.1
973.8	118.0	149.6	743.5	132.0
978.8	118.4	150.2	743.0	131.7
983.8	118.0	149.6	743.5	131.7
988.8	118.2	149.9	743.2	131.4
993.8	118.3	150.1	743.1	131.2
998.8	118.3	150.0	743.2	131.0
1003.8	118.1	149.7	743.4	130.9
1008.8	117.7	149.3	743.7	130.8
1013.8	117.8	149.4	743.6	130.6
1018.8	64.3	81.5	797.2	144.4
1198.8	61.9	78.5	799.5	144.4
1200.0	61.9	78.5	735.1	183.1

Table 6.4.1-11 (Cont.) Double-Ended Pump Suction Break Maximum Safeguards, Post-Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
828.8	119.6	151.6	741.9	132.6
833.8	119.1	151.0	742.4	132.6
838.8	118.6	150.4	742.8	132.5
843.8	119.3	151.2	742.2	132.2
848.8	118.8	150.6	742.7	132.2
853.8	119.4	151.4	742.1	131.8
858.8	118.9	150.7	742.6	131.8
863.8	119.4	151.4	742.0	131.5
868.8	118.9	150.8	742.6	131.5
873.8	119.4	151.4	742.0	131.2
878.8	118.9	150.7	742.6	131.2
883.8	119.3	151.3	742.1	130.9
888.8	118.7	150.6	742.7	130.9
893.8	119.2	151.1	742.3	130.6
898.8	118.5	150.3	742.9	130.6
903.8	118.9	150.8	742.6	130.4
908.8	118.3	149.9	743.2	130.4
913.8	118.6	150.3	742.9	130.2
918.8	118.8	150.7	742.6	133.8
923.8	119.1	151.0	742.4	133.6
928.8	118.3	150.0	743.1	133.6
933.8	118.5	150.2	743.0	133.3
938.8	118.6	150.4	742.8	133.1
943.8	118.7	150.5	742.8	132.9
948.8	118.7	150.5	742.8	132.7
953.8	118.7	150.5	742.8	132.6
958.8	118.6	150.4	742.9	132.4
963.8	118.5	150.2	743.0	132.3
968.8	118.3	149.9	743.2	132.1
973.8	118.0	149.6	743.5	132.0
978.8	118.4	150.2	743.0	131.7
983.8	118.0	149.6	743.5	131.7
988.8	118.2	149.9	743.2	131.4
993.8	118.3	150.1	743.1	131.2
998.8	118.3	150.0	743.2	131.0
1003.8	118.1	149.7	743.4	130.9
1008.8	117.7	149.3	743.7	130.8
1013.8	117.8	149.4	743.6	130.6
1018.8	64.3	81.5	797.2	144.4
1198.8	61.9	78.5	799.5	144.4
1200.0	61.9	78.5	735.1	183.1

Table 6.4.1-11 (Cont.) Double-Ended Pump Suction Break Maximum Safeguards, Post-Reflood M&E Releases				
	Break Path No. 1*		Break Path No. 2**	
Time (sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)	Flow (lbm/sec)	Energy (Thousand Btu/sec)
1283.4	61.9	78.5	735.1	183.1
1283.5	71.0	88.8	726.0	180.3
1285.0	70.9	88.8	726.0	180.3
1786.3	70.9	88.8	726.0	180.3
1786.4	65.0	74.8	731.9	121.8
3599.9	54.9	63.1	742.1	123.6
3600.1	43.4	49.9	753.6	112.4
18000.1	26.4	30.4	776.7	97.9
18001.1	26.5	30.5	775.3	101.6
30000.1	23.4	26.9	780.3	96.8
106400.1	16.0	18.4	790.4	89.4
1000000.1	6.9	7.9	802.8	79.6
2592000.1	4.7	5.4	805.1	79.5
10000000.0	2.2	2.5	807.6	79.7

* M&E exiting the SG side of the break

** M&E exiting the pump side of the break

Table 6.4.1-12 LOCA M&E Release Analysis Core Decay Heat Fraction	
Time (sec)	**Decay Heat Generation Rate (Btu/hr)**
10	0.052293
15	0.049034
20	0.047562
40	0.041504
60	0.038493
80	0.036410
100	0.034842
150	0.032180
200	0.030432
400	0.026664
600	0.024486
800	0.022943
1000	0.021722
1500	0.019483
2000	0.017903
4000	0.014386
6000	0.012684
8000	0.011645
10000	0.010916
15000	0.010130
20000	0.009368
40000	0.007784
60000	0.006976
80000	0.006439
100000	0.006034
150000	0.005336
200000	0.004859
400000	0.003781
600000	0.003212
800000	0.002844
1000000	0.002589

Table 6.4.1-13 Double-Ended Hot-Leg Break Mass Balance				
		Time (Sec)		
		0.00	20.20	20.20*
		Mass (Thousand lbm)		
Initial	In RCS and ACC	623.83	623.83	623.83
Added Mass	Pumped Injection	0.00	0.00	0.00
	Total Added	0.00	0.00	0.00
Total Available		623.83	623.83	623.83
Distribution	Reactor Coolant	426.13	61.54	61.62
	Accumulator	197.70	147.59	147.52
	Total Contents	623.83	209.13	209.13
Effluent	Break Flow	0.00	414.68	414.68
	ECCS Spill	0.00	0.00	0.00
	Total Effluent	0.00	414.68	414.68
Total Accountable**		623.83	623.81	623.81

* This time is the bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill.

** The difference between total available mass and total accountable mass at later times in the calculation reflect calculational error due to round off, time step changes, etc.

Table 6.4.1-14 Double-Ended Pump Suction Break Mass Balance Minimum Safeguards								
		Time (Sec)						
		0.00	21.40(1)	21.40(2)	216.6(3)	1247.1(4)	1805.2(5)	3600.0(6)
		Mass (Thousand lbm)						
Initial	In RCS & Accumulator	623.83	623.83	623.83	623.83	623.83	623.83	623.83
Added Mass	Pumped Injection	0.00	0.00	0.00	94.23	636.17	929.64	1836.43
	Total Added	0.00	0.00	0.00	94.23	636.17	929.64	1836.43
Total Available		623.83	623.83	623.83	718.06	1260.00	1553.47	2460.26
Distribution	Reactor Coolant	426.13	49.29	56.88	106.51	106.51	106.51	106.51
	Accumulator	197.70	146.24	138.65	0.00	0.00	0.00	0.00
	Total Contents	623.83	195.53	195.53	106.51	106.51	106.51	106.51
Effluent	Break Flow	0.00	428.28	428.28	602.88	1144.82	1438.29	2345.08
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	428.28	428.28	602.88	1144.82	1438.29	2345.08
Total Accountable*		623.83	623.83	623.83	709.39	1251.33	1544.80	2451.59

Notes:

(1) End of Blowdown

(2) Bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill.

(3) End of Reflood

(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure

(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure

(6) Time at which both SGs equilibrate to 14.7 psia

* The difference between total available mass and total accountable mass at later times in the calculation reflect calculational error due to round off, time step changes, etc.

Table 6.4.1-15
Double-Ended Pump Suction Break
Mass Balance
Maximum Safeguards

		Time (Sec)						
		.00	21.40(1)	21.40(2)	223.8(3)	1283.51(4)	1786.30(5)	3600.00(6)
		Mass (Thousand lbm)						
Initial	In RCS & Accumulator	623.83	623.83	623.83	623.83	623.83	623.83	623.83
Added Mass	Pumped Injection	.00	.00	.00	162.65	1070.16	1470.87	2916.35
	Total Added	.00	.00	.00	162.65	1070.16	1470.87	2916.35
Total Available		623.83	623.83	623.83	786.48	1693.99	2094.70	3540.18
Distribution	Reactor Coolant	426.13	49.29	56.88	106.91	106.91	106.91	106.91
	Accumulator	197.70	146.24	138.65	.00	.00	.00	.00
	Total Contents	623.83	195.53	195.53	106.91	106.91	106.91	106.91
Effluent	Break Flow	.00	428.28	428.28	670.90	1578.41	1979.08	3424.56
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	.00	428.28	428.28	670.90	1578.41	1979.08	3424.56
Total Accountable*		623.83	623.81	623.81	777.80	1685.31	2085.99	3531.47

Notes:

(1) End of Blowdown

(2) Bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill

(3) End of Reflood

(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure.

(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure.

(6) Time at which both SGs equilibrate to 14.7 psia.

* The difference between total available mass and total accountable mass at later times in the calculation reflect calculational error due to round off, time step changes, etc.

Table 6.4.1-16 Double-Ended Hot-Leg Break Energy Balance				
		Time (Sec)		
		0.00	20.20	20.20*
		Energy (Million Btu)		
Initial Energy	In RCS, Acc, SG	736.00	736.00	736.00
Added Energy	Pumped Injection	0.00	0.00	0.00
	Decay Heat	0.00	5.92	5.92
	Heat From Secondary	0.00	-1.84	-1.84
	Total Added	0.00	4.09	4.09
Total Available		736.00	740.08	740.08
Distribution	Reactor Coolant	254.23	13.12	13.13
	Accumulator	19.64	14.66	14.66
	Core Stored	21.51	9.19	9.19
	Primary Metal	125.65	117.89	117.89
	Secondary Metal	83.41	81.57	81.57
	Steam Generator	231.56	225.19	225.19
	Total Contents	736.00	461.61	461.61
Effluent	Break Flow	0.00	277.98	277.98
	ECCS Spill	0.00	0.00	0.00
	Total Effluent	0.00	277.98	277.98
Total Accountable**		736.00	739.59	739.59

* This time is the bottom of core recovery time, which is identical to the end of blowdown time due to the assumption of instantaneous refill.

** The difference between total available mass and total accountable mass at later times in the calculation reflect calculational error due to round off, time step changes, etc.

Table 6.4.1-17
Double-Ended Pump Suction Break
Energy Balance
Minimum Safeguards

		Time (Sec)						
		0.00	21.40(1)	21.40(2)	216.6(3)	1247.13(4)	1805.2(5)	3600.0(6)
		Energy (Million Btu)						
Initial Energy	In RCS, Acc, SG	736.00	736.00	736.00	736.00	736.00	736.00	736.00
Added Energy	Pumped Injection	0.00	0.00	0.00	8.76	59.16	86.45	209.72
	Decay Heat	0.00	5.78	5.78	25.27	96.12	127.31	212.66
	Heat From Secondary	0.00	-.35	-.35	-.35	4.51	6.34	6.34
	Total Added	0.00	5.42	5.42	33.68	159.79	220.10	428.72
Total Available		736.00	741.42	741.42	769.68	895.79	956.09	1164.72
Distribution	Reactor Coolant	254.23	10.77	11.53	27.92	27.92	27.92	27.92
	Accumulator	19.64	14.53	13.77	0.00	0.00	0.00	0.00
	Core Stored	21.51	12.21	12.21	3.91	3.74	3.49	2.71
	Primary Metal	125.65	118.82	118.82	99.71	62.84	52.91	41.33
	Secondary Metal	83.41	83.06	83.06	76.51	50.12	40.37	31.64
	Steam Generator	231.56	230.29	230.29	208.52	134.41	109.90	87.06
	Total Contents	736.00	469.69	469.69	416.56	279.03	234.59	190.66
Effluent	Break Flow	0.00	271.25	271.25	345.42	609.07	706.59	959.89
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	271.25	271.25	345.42	609.07	706.59	959.89
Total Accountable*		736.00	740.93	740.93	761.99	888.10	941.18	1150.55

Notes:

(1) End of Blowdown

(2) Bottom of core recovery time. This time is identical to the end of blowdown time due to the assumption of instantaneous refill.

(3) End of Reflood

(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure

(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure

(6) Time at which both SGs equilibrate to 14.7 psia

* The difference between total available energy and total accountable energy at later times in the calculation reflect calculational error due to round off, time step changes, etc.

Table 6.4.1-18
Double-Ended Pump Suction Break
Energy Balance, Maximum Safeguards

		Time (sec)						
		.00	21.40[1]	21.40[2]	223.8[3]	1283.51[4]	1786.30[5]	3600.00[6]
		Energy (Million Btu)						
Initial Energy	In RCS, Acc, SG	736.00	736.00	736.00	736.00	736.00	736.00	736.00
Added Energy	Pumped Injection	.00	.00	.00	15.13	103.25	163.01	378.56
	Decay Heat	.00	5.78	5.78	25.89	98.27	126.30	212.65
	Heat From Secondary	.00	-.35	-.35	-.35	4.65	6.22	6.22
	Total Added	.00	5.42	5.42	40.66	206.17	295.54	597.43
Total Available								
Distribution	Reactor Coolant	736.00	741.42	741.42	776.66	942.16	1031.53	1333.43
	Accumulator	254.23	10.77	11.53	28.01	28.01	28.01	28.01
	Core Stored	21.51	12.21	12.21	3.91	3.74	3.51	2.71
	Primary Metal	125.65	118.82	118.82	99.51	62.17	53.11	41.37
	Secondary Metal	83.41	83.06	83.06	76.67	49.38	40.48	31.68
	Steam Generator	231.56	230.29	230.29	208.93	132.41	110.07	87.03
	Total Contents	736.00	469.69	469.69	417.04	275.70	235.19	190.80
Effluent	Break Flow	.00	271.25	271.25	351.92	658.77	779.73	1127.39
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	271.25	271.25	351.92	658.77	779.73	1127.39
Total Accountable*		736.00	740.93	740.93	768.96	934.47	1014.91	1318.18

Notes to Table 6.4.1-18:

(1) End of Blowdown

(2) Bottom of core recovery time. This time is identical to the end of blowdown time due to the assumption of instantaneous refill.

(3) End of Reflood

(4) Time at which the Broken Loop SG equilibrates at the first intermediate pressure.

(5) Time at which the Intact Loop SG equilibrates at the second intermediate pressure.

(6) Time at which both SGs equilibrate to 14.7 psia.

* The difference between total available energy and total accountable energy at later times in the calculation reflect calculational error due to round off, time step changes, etc.

Table 6.4.1-19 Double-Ended Hot-Leg Break Sequence Of Events	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.1	Low Pressurizer Pressure SI Setpoint - 1714.3 psia reached by SATAN
12.7	Broken Loop Accumulator Begins Injecting Water
12.8	Intact Loop Accumulator Begins Injecting Water
20.20	End of Blowdown Phase

Table 6.4.1-20 Double-Ended Pump Suction Break Minimum Safeguards Sequence Of Events	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.4	Low Pressurizer Pressure SI Setpoint - 1714.3 psia reached by SATAN
14.1	Broken Loop Accumulator Begins Injecting Water
14.5	Intact Loop Accumulator Begins Injecting Water
21.4	End of Blowdown Phase
32.4	Safety Injection Begins
53.27	Broken Loop Accumulator Water Injection Ends
53.72	Intact Loop Accumulator Water Injection Ends
216.6	End of Reflood Phase
1247.1	Broken Loop SG Secondary Side pressure reaches the first intermediate pressure
1805.2	Intact Loop SG Secondary Side pressure reaches the second intermediate pressure
2210.0	Cold Leg Recirculation Begins
3600.0	Broken and Intact Loop SG pressure reaches equilibrium at 14.7 psia
3.2E+07	Transient Modeling Terminated

Table 6.4.1-21 Double-Ended Pump Suction Break Maximum Safeguards Sequence Of Events	
Time (sec)	**Event Description**
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.4	Low Pressurizer Pressure SI Setpoint - 1714.3 psia reached by SATAN
14.1	Broken Loop Accumulator Begins Injecting Water
14.5	Intact Loop Accumulator Begins Injecting Water
21.4	End of Blowdown Phase
32.4	Safety Injection Begins
53.87	Broken Loop Accumulator Water Injection Ends
54.32	Intact Loop Accumulator Water Injection Ends
223.8	End of Reflood Phase
1283.5	Broken Loop SG Secondary Side pressure reaches the first intermediate pressure
1786.3	Intact Loop SG Secondary Side pressure reaches the second intermediate pressure.
1200.0	Cold Leg Recirculation Begins
3600.0	Broken and Intact Loop SG pressure reach equilibrium at 14.7 psia
2.592E+06	Transient Modeling Terminated

6.4.2 Short-Term LOCA Mass and Energy Releases

6.4.2.1 Introduction

The short-term LOCA-related mass and energy releases are used as input to the subcompartment analyses, which are performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) accompanying a high-energy line pipe rupture within that subcompartment. The subcompartments evaluated include the steam generator compartment, the reactor cavity region, and the pressurizer compartment. For the SG compartment and the reactor cavity region, the fact that the HNP is approved for leak-before-break (LBB) was used to qualitatively demonstrate that any changes associated with the SGR/Uprating program are offset by the LBB benefit of using the smaller RCS nozzle breaks. This demonstrates that the current licensing bases for these subcompartments remain bounding. For the pressurizer compartment, the Reference 5 methodology was applied to calculate pressurizer spray line and surge line M&E releases. The results of this evaluation have been provided to CP&L for use in the pressurizer subcompartment evaluation. (See the BOP Licensing Report.)

6.4.2.2 Description of Analyses and Evaluations

The current licensing basis analyses for short-term LOCA mass and energy releases are presented in Section 6.2.1.2 of the FSAR (Reference 7). These mass and energy releases were generated with the Westinghouse 1975 mass and energy model (Reference 5) for the following breaks to support subsequent analyses of the reactor cavity, steam generator, and pressurizer compartments:

Case	Break Description
1.	150 in^2 Cold Leg Break (reactor cavity blowdown)
2.	150 in^2 Hot Leg Break (reactor cavity blowdown)
3.	Double-Ended Cold Leg Break
4.	Double-Ended Hot Leg Break
5.	Double-Ended Pump Suction Break
6.	Double-Ended Pressurizer Surge Line Break
7.	Pressurizer Spray Line Break

Mass and energy releases for these breaks are presented in HNP FSAR Tables 6.2-12 through 6.4-18.

A reanalysis was conducted to determine the effect of the SGR/Uprating on the short-term LOCA-related M&E releases that support subcompartment analyses discussed in Chapter 6.2.1.2 of the HNP FSAR (Reference 7). The HNP was licensed for LBB by Reference 8. Therefore, only breaks in the largest branch lines are analyzed, which are the pressurizer surge line and spray line breaks found in the HNP FSAR (cases 6 and 7 from above). The remaining FSAR

breaks (i.e., breaks in the main RCS piping, cases 1 through 5) have been eliminated by LBB and therefore, the M&E releases associated with these breaks would bound any RCS primary break considered under the LBB exemption. This evaluation addresses the impact of the SGR/Uprating and other relevant issues on the current licensing basis for the HNP.

The subcompartment analysis is performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) which accompanies a high-energy line pipe rupture within the subcompartment. The magnitude of the pressure differential across the walls is a function of several parameters, which include the blowdown M&E release rates, the subcompartment volume, vent areas, and vent flow behavior. The blowdown M&E release rates are affected by the initial RCS temperature conditions. Since short-term releases are linked directly to the critical mass flux, which increases with decreasing temperatures, the short-term LOCA releases would be expected to increase due to any reductions in RCS coolant temperature conditions. Short-term blowdown transients are characterized by a peak M&E release rate that occurs during a subcooled condition. Therefore, using lower temperatures, which maximizes the short-term LOCA M&E releases, data representative of the lowest inlet and outlet temperatures (with uncertainty subtracted) were used for the HNP SGR/Uprating analysis.

HNP has a temperature operating range for T_{avg} of 572°F to 588.2°F. For this evaluation, an RCS pressure of 2301 psia (2250 + 51 psi uncertainty), a vessel outlet temperature of 598.2°F, and a vessel/core inlet temperature of 530.6°F were considered for the uprating, which includes consideration of the lower end to the temperature operating range with a temperature uncertainty of -6°F.

Additionally, due to the short time period (0-3 seconds) for which these events are analyzed, the ECCS system is not modeled. Since the ECCS will not start in this short time period, single failures in the ECCS and Engineered Safeguards are not a concern and are not considered.

6.4.2.3 Acceptance Criteria

The NRC's NUREG-0800, Section 6.2.1.3, "Mass and Energy Release analysis for Postulated Loss-of-Coolant Accidents," subsection II, provides guidance on the NRC's expectations for what must be included in a LOCA mass and energy release calculation. The NRC has determined that the Westinghouse M&E models described in WCAP-8264-P-A, Rev. 1 (Reference 5) satisfy those expectations.

6.4.2.5 Results

The results of the pressurizer surge line and spray line break analyses are given in Tables 6.4.2-1 and 6.4.2-2. The methodology described in Reference 5 was used.

Per Reference 8, the HNP is approved for LBB. LBB eliminates the dynamic effects of postulated primary loop pipe ruptures from the design basis. This means that the current breaks (a double-ended circumferential rupture of the reactor coolant cold leg, hot leg, and the steam generator inlet nozzle, used for the SG compartments, and a 150 in^2 RV inlet break for the reactor cavity region) no longer have to be considered for the short-term effects. Since the RCS piping has been eliminated from consideration, the large branch nozzles must be considered for design verification. This includes the surge line, accumulator line, and the RHR line. These smaller breaks, which are outside the cavity region, would result in minimal asymmetric pressurization in the reactor cavity region. Additionally, compared to the large RCS double-ended ruptures, the differential loadings are significantly reduced. For example, the peak break compartment pressure can be reduced by a factor of greater than 2, and the peak differential across an adjacent wall can be reduced by a factor of greater than 3, if only the nozzle breaks are considered. Therefore, since the HNP is approved for LBB, the decrease in M&E releases associated with the smaller RCS nozzle breaks, as compared to the larger RCS pipe breaks, more than offsets any increased releases associated with the lower RCS temperatures as a result of the SGR/Uprating. The current licensing basis subcompartment analyses that consider breaks in the RCS remain bounding.

6.4.2.6 Conclusions

The short-term LOCA-related M&E releases discussed in Chapter 6.2 of the HNP FSAR have been reviewed to assess the effects associated with the SGR/Uprating project. New analyses were performed for the pressurizer surge line and spray line breaks. The results of these new analyses appear in Tables 6.4.2-1 and 6.4.2-2. The results of these analyses were provided for use in the pressurizer subcompartment structural analysis. (See the BOP Licensing Report.) Since the HNP is approved for LBB, the decrease in M&E releases associated with the smaller RCS nozzle breaks, as compared to the larger RCS pipe breaks, more than offsets any increased releases associated with the SGR/Uprating project. The current licensing basis subcompartment analyses that consider breaks in the primary loop RCS piping (i.e., steam generator subcompartments and reactor cavity region), therefore, remain bounding. Further, this analysis at 2912.4 MWt bounds operation at 2787.4 MWt with the Model Δ75 replacement steam generators, since the lower bound RCS temperatures occur at the uprated power of 2912.4 MWt, which results in higher short-term M&E releases.

Table 6.4.2-1 Short-Term Mass and Energy Release for a Double-Ended Pressurizer Surge Line Break			
Time (sec)	**Break Flow (lbm/sec)**	**Break Energy (Btu/sec)**	**Enthalpy (Btu/lbm)**
0.00000	0.0000000E+00	0.0000000E+00	0.00
0.00100	2.1103676E+03	1.3534502E+06	641.33
0.00904	9.4237946E+03	6.0692558E+06	644.04
0.01004	9.8820135E+03	6.3686254E+06	644.47
0.02004	1.3838759E+04	8.9630591E+06	647.68
0.03003	1.4863396E+04	9.6634563E+06	650.15
0.04003	1.6093160E+04	1.0502445E+07	652.60
0.05001	1.6159154E+04	1.0577799E+07	654.60
0.06002	1.6553060E+04	1.0864070E+07	656.32
0.07001	1.6655118E+04	1.0953752E+07	657.68
0.08006	1.6433660E+04	1.0835927E+07	659.37
0.09004	1.6416382E+04	1.0846517E+07	660.71
0.10002	1.6907343E+04	1.1180615E+07	661.29
0.20005	1.4938162E+04	1.0006279E+07	669.85
0.30001	1.4111884E+04	9.4848110E+06	672.12
0.40005	1.4341020E+04	9.6094731E+06	670.07
0.50001	1.4291815E+04	9.5529874E+06	668.42
0.60001	1.4319763E+04	9.5477666E+06	666.75
0.70001	1.4293723E+04	9.5143628E+06	665.63
0.80003	1.4305338E+04	9.5119696E+06	664.92
0.90017	1.4272873E+04	9.4888186E+06	664.81
1.00000	1.4151129E+04	9.4151449E+06	665.33
1.10016	1.3962764E+04	9.3022100E+06	666.22
1.20014	1.3846967E+04	9.2358276E+06	666.99
1.30006	1.3682495E+04	9.1379399E+06	667.86
1.40019	1.3597790E+04	9.0918136E+06	668.62
1.50001	1.3546720E+04	9.0685560E+06	669.43
1.60021	1.3469738E+04	9.0282535E+06	670.26
1.70011	1.3380283E+04	8.9786760E+06	671.04
1.90006	1.3193815E+04	8.8690636E+06	672.21
2.00049	1.3130329E+04	8.8308523E+06	672.55

Note: The tabulated energy releases should be increased 8.35% in order to bound operation at a T_{avg} of 572°F with a -6.0°F temperature uncertainty and at best estimate RCS flow.

Table 6.4.2-2
Short-Term Mass and Energy Release
for a Double-Ended Pressurizer Spray Line Break

Time (sec)	Mass Flow (lb/sec)	Energy Flow (Btu/sec)	Avg. Enthalpy (Btu/lb)
0.00000	0.0000000E+00	0.0000000E+00	0.00
0.01103	5.5518493E+03	3.2905004E+06	592.69
0.02003	5.7362822E+03	3.3924253E+06	591.40
0.03004	5.7848213E+03	3.4191326E+06	591.05
0.04012	5.8059940E+03	3.4305494E+06	590.86
0.05008	5.6819445E+03	3.3609717E+06	591.52
0.06007	5.6565506E+03	3.3467035E+06	591.65
0.07002	5.7317091E+03	3.3884934E+06	591.18
0.08006	5.7132020E+03	3.3780441E+06	591.27
0.09004	5.6174032E+03	3.3245990E+06	591.84
0.10006	5.5866929E+03	3.3074862E+06	592.03
0.20007	5.5623287E+03	3.2933991E+06	592.09
0.30000	5.4512179E+03	3.2308583E+06	592.69
0.40009	5.3230373E+03	3.1592006E+06	593.50
0.50001	5.2181352E+03	3.1007846E+06	594.23
0.60009	5.1229225E+03	3.0478652E+06	594.95
0.70015	5.0239055E+03	2.9929317E+06	595.74
0.80014	4.9453679E+03	2.9493564E+06	596.39
0.90018	4.8895171E+03	2.9183478E+06	596.86
1.00013	4.9513805E+03	2.9521263E+06	596.22
1.10003	4.9736299E+03	2.9641328E+06	595.97
1.20003	5.0045052E+03	2.9809611E+06	595.66
1.30005	5.0311580E+03	2.9955387E+06	595.40
1.40002	5.0433071E+03	3.0021727E+06	595.28
1.50009	5.0585074E+03	3.0105403E+06	595.14
1.60003	5.0654476E+03	3.0143923E+06	595.09
1.70006	5.0667942E+03	3.0151892E+06	595.09
1.80003	5.0675444E+03	3.0156876E+06	595.10
1.90011	5.0613076E+03	3.0123487E+06	595.17
2.00022	5.0532974E+03	3.0080471E+06	595.26

Note: The tabulated energy releases should be increased 0.4% in order to bound operation at a T_{avg} of 572°F with a -6.0°F temperature uncertainty.

6.4.3 References

1. WCAP-10325-P-A, (Proprietary), WCAP-10326-A (Nonproprietary), "Westinghouse LOCA Mass & Energy Release Model for Containment Design - March 1979 Version," May 1983.

2. NUREG-1038, "Safety Evaluation Report Related to the Operation of Shearon Harris Nuclear Plant, Units 1 and 2," Section 6.2.1.3, November 1983.

3. Docket No. 50-315, "Amendment No. 126, Facility Operating License No. DPR-58 (TAC No. 7106), for D. C. Cook Nuclear Plant Unit 1," June 9, 1989.

4. EPRI 294-2, "Mixing of Emergency Core Cooling Water with Steam; 1/3-Scale Test and Summary," (WCAP-8423), Final Report, June 1975.

5. WCAP-8264-P-A, Rev. 1 (Proprietary), WCAP-8312-A (Nonproprietary), "Westinghouse Mass and Energy Release Data For Containment Design," August 1975.

6. ANSI/ANS-5.1 1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

7. Shearon Harris Nuclear Power Plant Final Safety Analysis Report, Section 6.2.

8. NRC Letter from G. W. Knighton to E. E. Utley (CP&L), "Request for Exemption from a portion of General Design Criteria 4 of Appendix A to 10 CFR Part 50 Regarding the need to Analyze Large Primary Loop Pipe Ruptures as Structural Design Basis for Shearon Harris Nuclear Power Plant Unit 1," June 5, 1985.

6.5 Main Steamline Break Mass and Energy Releases

6.5.1 Main Steamline Break Mass and Energy Releases Inside Containment

6.5.1.1 Introduction

Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment and elevated containment temperatures and pressures. The quantitative nature of the releases following a steamline rupture is dependent upon the plant initial operating conditions and the size of the rupture as well as the configuration of the plant steam system and the containment design. These variations make it difficult to determine the absolute worst cases for either containment pressure or temperature evaluation following a steamline break. The analysis considers a variety of postulated pipe breaks encompassing wide variations in plant operation, safety system performance, and break size in determining the main steamline break (MSLB) mass and energy (M&E) releases for use in containment integrity analysis.

6.5.1.2 Description of Analyses

The description of the analysis and methods pertaining to the main steamline break mass and energy releases inside containment are presented below.

To determine the effects of plant power level and break area on the mass and energy releases from a ruptured steamline, spectra of both variables have been evaluated. At plant power levels of 102 percent, 70 percent, 30 percent and 0 percent of nominal full-load power, two break sizes have been defined. These break areas are defined as the following.

1. A full double-ended guillotine break (DEGB) downstream of the flow restrictor in one steamline. Note that a DEGB is defined as a rupture in which the steam pipe is completely severed and the ends of the break displace from each other. The full DEGB represents the largest break of the main steamline producing the highest mass flowrate from the faulted-loop steam generator.

2. A small split rupture that will neither generate a steamline isolation signal from the Westinghouse Solid-State Protection System (SSPS) nor result in water entrainment in the break effluent. Reactor protection and safety injection actuation functions are obtained from containment pressure signals.

The 12 cases included in the Harris Nuclear Plant (HNP) SGR/Uprating analysis have been chosen based on the results of the analyses presented in the HNP FSAR, subsection 6.2.1.4. The cases, listed in subsection 6.5.1.2.16 of this licensing report, have been analyzed assuming operation with the Westinghouse-design Model Δ75 replacement steam generators at the uprated

power condition. Other important plant conditions and features are discussed in the following paragraphs.

6.5.1.2.1 Initial Power Level

Steamline breaks can be postulated to occur with the plant in any operating condition ranging from hot shutdown to full power. Since steam generator mass decreases with increasing power level, breaks occurring at lower power levels will generally result in a greater total mass release to the containment. However, because of increased stored energy in the primary side of the plant, increased heat transfer in the steam generators, and additional energy generation in the fuel, the energy release to the containment from breaks postulated to occur during full-power, or near full-power, operation may be greater than for breaks occurring with the plant in a low-power, or hot-shutdown, condition. Additionally, pressure in the steam generators changes with increasing power and has a significant influence on the rate of blowdown.

Because of the opposing effects on mass versus energy release for the MSLB due to a change in initial power level, a single power level cannot be specified as the worst case for either the containment pressure cases or the containment temperature cases. Therefore, representative power levels including 102 percent, 70 percent, 30 percent and 0 percent of nominal full-power conditions have been investigated for HNP as presented in the HNP FSAR, based on the information in Reference 1. Reference 1 has been reviewed and approved by the NRC for use in MSLB analysis inside containment. Additional discussion is provided in subsection 6.5.1.2.16 of this report.

In general, the plant initial conditions are assumed to be at the nominal value corresponding to the initial power for that case, with appropriate uncertainties included. Tables 6.5.1-1 and 6.5.1-2 identify the values assumed for RCS pressure, RCS vessel average temperature, RCS flow, pressurizer water volume, steam generator water level, steam generator pressure, and feedwater enthalpy corresponding to each power level analyzed. Steamline break mass and energy releases assuming an RCS average temperature at the high end of the T_{avg} window are conservative with respect to similar releases at the low end of the T_{avg} window. At the high end, there is more mass and energy available for release into containment. The thermal design flowrate has been used for the RCS flow input consistent with the assumptions documented in Reference 1. The thermal design flowrate is also consistent with other MSLB analysis assumptions related to nonstatistical treatment of uncertainties, as well as RCS thermal-hydraulic inputs related to pressure drops and rod drop time.

Uncertainties on the initial conditions assumed in the analysis for the SGR/Uprating program have been applied only to the RCS average temperature (6°F), the steam generator mass (8 percent narrow-range span), and the power fraction (2 percent) and feedwater enthalpy (2°F) at full power. Nominal values are adequate for the initial conditions associated with pressurizer pressure and pressurizer water level. Uncertainty conditions are only applied to those parameters that could increase the amount of mass or energy discharged into containment.

6.5.1.2.2 Single-Failure Assumptions

To avoid unnecessary conservatism, bounding multiple failure assumptions have not been made in the analysis. Each case analyzed considered only one single failure. One of these failures results in minimum containment spray and fan coolers to allow for a failure of a train of containment safeguards features. The following single failures are postulated (discussed also in Reference 1), which may significantly affect the containment results.

a. Failure of the Main Steam Isolation Valve (MSIV) in the Faulted Loop

The main steamline isolation function is accomplished via the MSIV in each of the three steamlines. Each valve closes on an isolation signal to terminate steam flow from the associated steam generator. The main steamline rupture upstream of this valve, as postulated for the inside-containment analysis, creates a situation in which the steam generator on the faulted loop cannot be isolated, even when the MSIV successfully closes. The break location allows a continued blowdown from the faulted-loop steam generator until it is empty and all sources of feedwater and auxiliary feedwater addition are terminated. If the faulted-loop MSIV fails to close, blowdown from more than one steam generator is prevented by the closure of the corresponding MSIV for each intact-loop steam generator. Therefore, there is no failure of a single MSIV that could cause continued blowdown from multiple steam generators.

In addition to the continued blowdown from the faulted-loop steam generator after MSIV closure, the steam in the unisolable section of the steamline needs to be considered. An MSIV failure can impact the mass and energy releases, since a failed MSIV will result in a larger unisolable steamline volume. The analytical method of addressing the steamline piping blowdown and the effect of an MSIV failure is dependent on break type, as discussed in subsection 6.5.1.2.16.

b. Failure of the Main Feedwater Isolation Valve (MFIV) in the Faulted Loop

If the MFIV in the feedwater line to the faulted steam generator is assumed to fail in the open position, backup isolation is provided via the main feedwater flow control valve (MFCV) closure. The additional inventory between the MFIV and the MFCV in the faulted loop would be available to be released to containment. (The effects of this single-failure assumption are being addressed separately in the BOP Licensing Report for the steam generator replacement and power uprate analysis.)

c. Failure of the Emergency Diesel Generator (EDG)

For the circumstance where offsite power is lost, power will be lost to the reactor coolant pumps (RCPs) and the EDGs will be relied upon to supply emergency power to the safeguards equipment. If one EDG fails in this situation, one train of safety injection (SI)

as well as one train of the containment safeguards functions will be lost. The only effect on the mass and energy releases is the loss of one SI train, a longer delay until SI actuation, and RCP trip. As noted later, minimum SI flow is assumed for all cases. The effect of reduced containment safeguards is accounted for in the containment response analysis. The assumption of a trip of all the RCPs coincident with reactor trip is less limiting than with offsite power available since the mass and energy releases are reduced due to the loss of forced reactor coolant flow, resulting in less primary-to-secondary heat transfer. Therefore, all MSLB M&E release cases are analyzed with the RCPs continuing to operate.

6.5.1.2.3 Main Feedwater System

The rapid depressurization that occurs following a steamline rupture typically results in large amounts of water being added to the steam generators through the main feedwater system. Rapid-closing MFIVs or MFCVs in the main feedwater lines limit this effect. The feedwater addition that occurs prior to closing of the MFIVs or MFCVs influences the steam generator blowdown in several ways. First, because the water entering the steam generator is subcooled, it lowers the steam pressure thereby reducing the flowrate out of the break. As the steam generator pressure decreases, some of the fluid in the feedwater lines downstream of the control valves will flash into the steam generators providing additional secondary fluid which may exit out of the rupture. Secondly, the increased flow causes an increase in the total heat transfer from the primary to secondary systems resulting in greater integrated energy being released out of the break.

Following the initiation of the MSLB, main feedwater flow is conservatively modeled by assuming that sufficient feedwater flow is provided to match or exceed the steam flow prior to reactor trip. The initial increase in feedwater flow (until fully isolated) is in response to the feedwater control valve opening up in response to the steam flow/feedwater flow mismatch, or the decreasing steam generator water level as well as due to a lower backpressure on the feedwater pump as a result of the depressurizing steam generator. This maximizes the total mass addition prior to feedwater isolation. The feedwater isolation response time, following the safety injection signal, is assumed to be a total of 10 seconds, accounting for delays associated with signal processing plus MFIV stroke time. (At zero-power initial conditions, the feedwater isolation response time, following the safety injection signal, is assumed to be a total of 12 seconds.) (For the circumstance in which the MFIV in the faulted loop fails to close, the effects of the feedwater isolation response time are being addressed separately in the BOP Licensing Report for the SGR/Uprating program.)

Following feedwater isolation, as the steam generator pressure decreases, some of the fluid in the feedwater lines downstream of the isolation valve may flash to steam if the feedwater temperature exceeds the saturation temperature. This unisolable feedwater line volume is an additional source of fluid that can increase the mass discharged out of the break. The unisolable volume in the feedwater lines is maximized for the faulted loop.

For the circumstance in which the MFIV in the faulted loop fails to close, the effects of the increase in the unisolable feedwater line volume are being addressed separately in the BOP Licensing Report for the SGR/Uprating program.

Steamline break mass and energy releases assuming a main feedwater temperature at the high end of the feedwater temperature window are conservative with respect to similar releases at the low end of the feedwater temperature window. At the high end, there is more energy available for release into containment.

6.5.1.2.4 Auxiliary Feedwater System

Generally, within the first minute following a steamline break, the auxiliary feedwater (AFW) system is initiated on any one of several protection system signals. Addition of auxiliary feedwater to the steam generators will increase the secondary mass available for release to containment as well as increase the heat transferred to the secondary fluid. The auxiliary feedwater flow to the faulted and intact steam generators has been assumed to be a constant value in the steamline break analysis inside containment. A high AFW flowrate to the faulted-loop steam generator is conservative for the steamline break event; therefore, these flows have been maximized. Conversely, a low AFW flowrate is conservative for the intact-loop steam generators; thus, these flows have been minimized. The volume of the AFW piping is minimized. Purging of AFW piping is not assumed since a minimum volume permits colder AFW to be injected into the steam generator rather than any hotter auxiliary feedwater resident in the piping. The more dense injected AFW causes a greater mass addition to the faulted-loop steam generator than if the resident auxiliary feedwater had to be purged prior to the flow of AFW into the steam generator. Auxiliary feedwater flow to the faulted-loop steam generator has been assumed up until the time of automatic auxiliary feedwater isolation on a steamline ΔP signal. Flashing of water resident in the AFW piping after isolation will not occur since the fluid temperature is less than the saturation temperature for containment pressure conditions. After isolation, auxiliary feedwater flow continues to intact-loop steam generators.

6.5.1.2.5 Steam Generator Fluid Mass

A maximum initial steam generator mass in the faulted-loop steam generator has been used in all of the analyzed cases. The use of a high faulted-loop initial steam generator mass maximizes the steam generator inventory available for release to containment. The initial mass has been calculated as the value corresponding to the programmed level +8 percent narrow-range span. Minimum initial masses in the intact-loop steam generators have been used in all of the analyzed cases. The use of reduced initial steam generator masses minimizes the availability of the heat sink afforded by the steam generators on those loops. The initial masses have been calculated as the value corresponding to the programmed level -8 percent narrow-range span. Steam generator reverse heat transfer is discussed in the following subsection.

All steam generator fluid masses are calculated corresponding to 0 percent tube plugging, which is conservative with respect to maximizing the fluid inventory and the primary-to-secondary heat transfer through the faulted-loop steam generator resulting from the steamline break.

6.5.1.2.6 Steam Generator Reverse Heat Transfer

Once the steamline isolation is complete, the steam generators in the intact loops become sources of energy that can be transferred to the steam generator with the broken steamline. This energy transfer occurs via the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the steam generator tubes could drop below the temperature of the secondary fluid in the intact steam generators, resulting in energy being returned to the primary coolant. This energy is then available to be transferred to the steam generator with the broken steamline. When applicable, the effects of reverse steam generator heat transfer are included in the results.

6.5.1.2.7 Break Flow Model

Piping discharge resistances are not included in the calculation of the releases resulting from the steamline ruptures [Moody Curve for an $f(l/D) = 0$ is used]. This is consistent with the expectations of the NRC as presented in Section 6.2.1.4 of the Standard Review Plan. For the HNP analysis, no entrainment is assumed in the break effluent. The assumption of saturated steam being released for all break types is a conservative assumption that maximizes the energy release into containment.

6.5.1.2.8 Steamline Volume Blowdown

The contribution to the mass and energy releases from the steam in the secondary plant main steam loop piping and header has been included in the mass and energy release calculations. The initial flowrate is determined using the Moody correlation, the pipe cross-sectional area, and the initial steam pressure. This blowdown is calculated only for the DEGB steamline break.

For the split-rupture steamline break, the unisolable steam mass in the piping is included as part of the initial inventory in the faulted-loop steam generator since the break is not large enough to cause a sudden decompression of the piping.

The analytical method of addressing the steamline piping blowdown and the effect of an MSIV failure or no MSIV failure is discussed in subsection 6.5.1.2.16.

6.5.1.2.9 Main Steamline Isolation

Steamline isolation is assumed in all three loops to terminate the blowdown from the two intact steam generators. A delay time of 7 seconds, accounting for delays associated with signal processing plus MSIV stroke time, with unrestricted steam flow through the valve during the valve stroke, has been assumed.

6.5.1.2.10 Protection System Actuations

The protection systems available to mitigate the effects of a MSLB accident inside containment include reactor trip, safety injection, steamline isolation, and feedwater isolation. (Analysis of the containment response to the MSLB is presented in the BOP Licensing Report.) The protection system actuation signals and associated setpoints that have been modeled in the analysis are identified in Table 6.5.1-3. The setpoints used are conservative values with respect to the plant-specific values delineated in the HNP Technical Specifications for Uprate conditions.

For the double-ended rupture MSLB at all power levels, the first protection system signal actuated is Low Steamline Pressure (lead/lag compensated in each channel) in any loop that initiates steamline isolation and safety injection; the safety injection signal produces a reactor trip signal. Feedwater system isolation occurs as a result of the safety injection signal.

For the split-rupture steamline breaks at all power levels, no mitigation signals are received from either the Reactor Protection System or any secondary-side signals produced by the Engineered Safety Features Actuation System. The first protection system signal actuated is assumed to be the High Containment Pressure which initiates safety injection; the safety injection signal produces a reactor trip signal. Feedwater system isolation occurs as a result of the safety injection signal. Steamline isolation is initiated following receipt of the High-High Containment Pressure signal.

The turbine stop valve is assumed to close instantly following the reactor trip signal.

6.5.1.2.11 Safety Injection System

Minimum safety injection system (SIS) flowrates corresponding to the failure of one SIS train have been assumed in this analysis. A minimum SI flow is conservative since the reduced boron addition maximizes a return to power resulting from the RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing steam flow out of the break. The delay time to achieve full SI flow is assumed to be 27 seconds for this analysis with offsite power available. A coincident loss of offsite power is not assumed for the analysis of the steamline break inside containment since the mass and energy releases would be reduced due to the loss of forced reactor coolant flow, resulting in less primary-to-secondary heat transfer.

6.5.1.2.12 Reactor Coolant System Metal Heat Capacity

As the primary side of the plant cools, the temperature of the reactor coolant could drop below the temperature of the reactor coolant piping, the reactor vessel, the reactor coolant pumps, and the steam generator thick-metal mass and tubing. As this occurs, the heat stored in the metal is available to be transferred to the steam generator with the broken line. Stored metal heat, however, does not have a major impact on the calculated mass and energy releases. The effects

of this RCS metal heat are included in the results using conservative thick-metal masses and heat transfer coefficients.

6.5.1.2.13 Core Decay Heat

Core decay heat generation assumed in calculating the steamline break mass and energy releases is based on the 1979 ANS Decay Heat + 2σ model (Reference 2). The existing analysis assumed the use of the 1971 standard (+20 percent uncertainty) for the decay heat. The assumption of using the 1979 version represents a deviation from the current licensing-basis analysis for HNP. This version of the decay heat input has been applied previously to other HNP licensing-basis safety analyses (FSAR Chapter 15, Reference 15.0.10-3).

6.5.1.2.14 Rod Control

The rod control system is conservatively assumed to be in manual operation for all steamline break analyses.

6.5.1.2.15 Core Reactivity Coefficients

Conservative core reactivity coefficients corresponding to end-of-cycle conditions, including hot zero power (HZP) stuck-rod moderator density coefficients, are used to maximize the reactivity feedback effects resulting from the steamline break. Use of maximum reactivity feedback results in higher power generation if the reactor returns to criticality, thus maximizing heat transfer to the secondary side of the steam generators.

6.5.1.2.16 Description of Analysis Cases

The system transient that provides the break flows and enthalpies of the steam release through the steamline break inside containment has been analyzed with the LOFTRAN (Reference 3) computer code. Blowdown mass and energy releases determined using LOFTRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant system thick-metal heat storage including steam generator thick-metal mass and tubing, and reverse steam generator heat transfer. As noted in Section 6.5.1.2.7, no entrainment is assumed in the break effluent for the HNP analysis. The assumption of saturated steam being released for all break types is a conservative assumption that maximizes the energy release into containment. This does not reflect a deviation in the HNP licensing basis since the current MSLB mass and energy releases analysis does not assume entrainment.

The existing MSLB M&E analysis inside containment was performed using the MARVEL code as documented in WCAP-8822. The use of the LOFTRAN code for the analysis of the MSLB M&E releases is documented in Supplement 1 of WCAP-8822 (Reference 1) and has been reviewed and approved by the NRC for this application. The LOFTRAN code has been utilized previously for the HNP licensing-basis safety analyses.

The HNP NSSS has been analyzed to determine the transient steam mass and energy releases inside containment following a steamline break event. Superheated steam is not assumed in the mass and energy releases from the faulted-loop steam generator for this analysis. The approved methodology for the MSLB M&E releases inside containment as documented in Supplement 2 of WCAP-8822 (Reference 1) does not require that the effects of steam superheat be considered for input to an analysis assuming a large, dry containment. Since the HNP containment design is of this type, the steam superheat assumption has not been considered in this analysis for the steam generator replacement and power uprate program. The resulting tables of mass and energy releases are used as input conditions to the analysis of the containment response.

The following licensing-basis cases of the MSLB inside containment have been analyzed at the noted conditions for the SGR/Uprating program.

Case 1: Full double-ended (1.4 ft^2) rupture at 102 percent power

Case 2: 0.687 ft^2 split rupture at 102 percent power – no single failure in MSLB transient

Case 3: 0.687 ft^2 split rupture at 102 percent power – MSIV failure

Case 4: Full double-ended (1.4 ft^2) rupture at 70 percent power

Case 5: 0.675 ft^2 split rupture at 70 percent power – no single failure in MSLB transient

Case 6: 0.675 ft^2 split rupture at 70 percent power – MSIV failure

Case 7: Full double-ended (1.4 ft^2) rupture at 30 percent power

Case 8: 0.666 ft^2 split rupture at 30 percent power – no single failure in MSLB transient

Case 9: 0.666 ft^2 split rupture at 30 percent power – MSIV failure

Case 10: Full double-ended (1.4 ft^2) rupture at hot standby (0 percent power)

Case 11: 0.558 ft^2 split rupture at 0 percent power – no single failure in MSLB transient

Case 12: 0.558 ft^2 split rupture at 0 percent power – MSIV failure

For the double-ended rupture cases, the forward-flow cross-sectional area from the faulted-loop steam generator is limited by the integral flow restrictor area of 1.4 ft^2, which is less than the actual area of 4.9 ft^2 for the main steam piping inside containment. The cross-sectional area of the steam piping at this location is larger than the sum of the flow restrictors in the intact-loop steam generators. Therefore, the larger cross-sectional area of the ruptured steamline expels steam faster than the smaller cross-sectional area of the intact-loop steam generator flow restrictors can fill it. Thus, the blowdown of the initial steam in the steamline header piping is modeled in the first few seconds of the event, followed by the reverse-flow blowdown from the intact-loop steam generators until MSIV closure. The initial reverse-flow blowdown is discussed in subsection 6.5.1.2.8, Steamline Volume Blowdown, and provided in Table 6.5.1-4.

The full DEGB represents the break producing the highest mass flowrate from the faulted-loop steam generator. Smaller DEGB break sizes are represented by a reduction in the initial steam blowdown rate at the time of the break. Therefore, no other DEGB break sizes have been considered other than the full DEGB.

For the split-break MSLB cases, the break area is smaller than the area of a single integral flow restrictor. The flowrate from all steam generators prior to MSIV closure and the flowrate from a single steam generator after MSIV closure supply the steam flow to the break. The steam in the unisolable portion of the steamline does not affect the blowdown until the time of steam generator dry out, when the flowrate from the steam generator would decrease below the critical flowrate out of the break. At this point, the additional steam in the piping begins to have an effect on break flowrate until the steamline piping is empty. To model this effect in LOFTRAN, the mass of the unisolable steam in the steamline is added to the initial mass of the faulted steam generator. This accurately reflects both the total mass and energy that will be released from the break, and the timing of the effect of the unisolable steamline volume on the blowdown.

All the cross-sectional split-rupture areas have been redefined based on the assumption of operation with the Westinghouse-design Model Δ75 replacement steam generators. Each break size as a function of power is the largest area that does not produce a steamline isolation signal from the Westinghouse SSPS, nor result in water entrainment in the break effluent as discussed in Reference 1.

6.5.1.3 Acceptance Criteria

The main steamline break is classified as an ANS Condition IV event, an infrequent fault. The acceptance criteria associated with the steamline break event resulting in a mass and energy release inside containment is based on an analysis that provides sufficient conservatism to assure that the containment design margin is maintained. The specific criteria applicable to this analysis are related to the assumptions regarding power level, stored energy, the break flow model, main and auxiliary feedwater flow, steamline and feedwater isolation, and single failure such that the containment peak pressure and temperature are maximized. These analysis assumptions have been included in this steamline break mass and energy release analysis as discussed in Reference 1 and subsection 6.5.1.2 of this report. The tables of mass and energy release data for each of the steamline break cases noted in the previous section are used as input to a containment response calculation to confirm the design parameters of the HNP containment structure.

6.5.1.4 Results

Using Reference 1 as a basis, including parameter changes associated with the Model Δ75 replacement steam generators and power uprating, the mass and energy release rates for each of the steamline break cases noted in subsection 6.5.1.2.16 have been developed for use in containment pressure and temperature response analyses. Tables 6.5.1-5 and 6.5.1-6 provide the sequence of events for the two limiting steamline breaks inside containment. Note that for cases

that credit the High-1 and High-2 containment pressure signals, the actuation times were provided informally based on preliminary (shortened) transients. The final split-rupture cases reflect actuation times that are conservative with respect to the preliminary actuation times.

6.5.1.5 Conclusions

The mass and energy releases from the 12 steamline break cases have been analyzed at the conditions defined by the steam generator replacement and uprated power level. The analysis methods delineated in subsection 6.5.1.2 have been included in the steamline break analysis such that the results are consistent with and continue to comply with the current HNP licensing-basis/acceptance requirements. The 102-percent-power and part-power MSLB M&E releases inside containment obtained with the Model Δ75 replacement steam generators at the uprated power of 2912.4 MWt were generated using a method consistent with MSLB M&E releases that would be obtained with the Model Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt. The steam mass and energy releases discussed in this section have been provided for use in the containment response analysis in support of the HNP SGR/Uprating program. The results of the containment response analysis are provided in the BOP Licensing Report.

6.5.1.6 References

1. WCAP-8822 (Proprietary) and WCAP-8860 (Nonproprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-S1-A (Nonproprietary), "Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP-8860-S2-A (Nonproprietary), "Supplement 2 – Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs," September 1986.

2. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

3. WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), "LOFTRAN Code Description," April 1984.

Table 6.5.1-1 Nominal Plant Parameters for SGR/Uprating* (MSLB M&E Releases Inside Containment)	
Nominal Conditions	
NSSS Power, MWt	2912.4
Core Power, MWt	2900
Reactor Coolant Pump Heat, MWt	12.4
Reactor Coolant Flow (total), gpm (Thermal Design Flow)	277,800
Pressurizer Pressure, psia	2250
Core Bypass, %	7.1
Reactor Coolant Vessel Average Temperature, °F	588.8*
Steam Generator**	
Steam Temperature, °F	545.9
Steam Pressure, psia	1011
Steam Flow, 10^6 lbm/hr (Plant Total)	12.87
Feedwater Temperature, °F	440
Zero-Load Temperature, °F	557

* Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T_{avg} window.

** Steam generator performance data used in the analysis is conservatively high for steam temperature and pressure. This data also corresponds to best-estimate RCS flow conditions.

Table 6.5.1-2 Initial Condition Assumptions for SGR/Uprating* (MSLB M&E Releases Inside Containment)				
Initial Conditions	**Power Level (%)**			
Parameter	**102**	**70**	**30**	**0**
RCS Average Temperature (°F)	594.8*	585.3*	572.5*	557.0
RCS Flowrate (gpm) (Thermal Design Flow)	277,800	277,800	277,800	277,800
RCS Pressure (psia)	2250	2250	2250	2250
Pressurizer Water Volume (ft^3)	836.5	704.9	529.4	397.8
Feedwater Enthalpy (Btu/lbm)	421.7	371.7	302.3	90.6
SG Pressure (psia)**	1059	1082	1115	1093
SG Water Level, faulted/intact (% NRS)	66.3/48.7	66.3/48.7	66.3/48.7	66.3/48.7

* Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T_{avg} window; temperatures include applicable calorimetric uncertainties.

** The noted SG pressures are determined at the steady-state conditions defined by the RCS average temperatures, including applicable uncertainties except at 0% power.

Table 6.5.1-3 **Protection System Actuation Signals and Safety System Setpoints for SGR/Uprating** **(MSLB M&E Releases Inside Containment)**	
<u>Reactor Trip</u> 2/3 Low Pressurizer Pressure – 1935 psia Safety Injection	
<u>Safety Injection</u> 2/3 Low Pressurizer Pressure – 1714.3 psia 2/3 Low Steamline Pressure in any loop – 556.9 psia dynamic compensation lead - 50 seconds lag - 5 seconds 2/3 High Containment Pressure - *	(implicit - used in the containment response analysis)
<u>Steamline Isolation</u> 2/3 Low Steamline Pressure in any loop – 556.9 psia dynamic compensation lead - 50 seconds lag - 5 seconds 2/3 High-High Containment Pressure - *	(implicit - used in the containment response analysis)
<u>Feedwater Isolation and Auxiliary Feedwater Initiation</u> Safety Injection	

* Setpoint is not explicitly modeled in MSLB M&E release analyses.

Table 6.5.1-4 Mass and Energy Flowrates for Steam Piping Reverse Flow Blowdown – Applicable to the DEGB MSLBs Inside Containment				
Power Level	102%	70%	30%	0%
Steam Mass Flowrate (lbm/sec)	10,743.8	11,022.3	11,398.9	11,162.0
Steam Energy Flowrate (10^6 Btu/sec)	12.795	13.115	13.547	13.275
Duration of Blowdown (sec)	2.084	2.088	2.094	2.090

Table 6.5.1-5 Sequence of Events (Peak Temperature Case)	
Time (sec)	**Event Description**
0.0	Main Steamline Break Occurs
	Low Steamline Pressure Setpoint Reached
	Rod Motion Starts (Low Steamline Pressure actuates SI which initiates Reactor Trip)
	High Containment Pressure Setpoint Reached
	Feedwater Isolation Occurs
	Steamline Isolation Occurs (following receipt of the Low Steamline Pressure signal)
	High-High Containment Pressure Setpoint Reached
	Safety Injection Flow Initiated
	Containment Fan Cooler Actuates
	Containment Spray Actuates
	Peak Containment Temperature Occurs
	Peak Containment Pressure Occurs
	Mass and Energy Releases Terminate (SG Dryout)

(IT IS THE RESPONSIBILITY OF CP&L TO ENSURE THAT THE INFORMATION IN THIS TABLE REFLECTS THE MOST-LIMITING MSLB CASE REPRESENTING THE PEAK TEMPERATURE CONTAINMENT RESPONSE.)

Table 6.5.1-6 Sequence of Events (Peak Pressure Case)	
Time (sec)	**Event Description**
0.0	Main Steamline Break Occurs
	High Containment Pressure Setpoint Reached
	Rod Motion Starts (High Containment Pressure actuates SI which initiates Reactor Trip)
	Feedwater Isolation Occurs
	Safety Injection Flow Initiated
	High-High Containment Pressure Setpoint Reached
	Steamline Isolation Occurs (following receipt of the High-High Containment Pressure signal)
	Containment Fan Cooler Actuates (following receipt of the High Containment Pressure signal)
	Containment Spray Actuates
	Peak Containment Temperature Occurs
	Peak Containment Pressure Occurs
	Mass and Energy Releases Terminate (SG Dryout)

(IT IS THE RESPONSIBILITY OF CP&L TO ENSURE THAT THE INFORMATION IN THIS TABLE REFLECTS THE MOST-LIMITING MSLB CASE REPRESENTING THE PEAK PRESSURE CONTAINMENT RESPONSE.)

6.5.2 Main Steamline Break Mass and Energy Releases Outside Containment

6.5.2.1 Introduction

Steamline ruptures occurring outside the reactor containment structure may result in significant releases of high-energy fluid to the structures surrounding the steam systems. Superheated steam blowdowns following the steamline break have the potential to raise compartment temperatures outside containment. Early uncovery of the steam generator tube bundle maximizes the enthalpy of the superheated steam releases out the break. The impact of the steam releases depends on the plant configuration at the time of the break, the plant response to the break, as well as the size and location of the break. Because of the interrelationship among many of the factors that influence steamline break mass and energy releases, an appropriate determination of a single limiting case with respect to mass and energy releases cannot be made. Therefore, it is necessary to analyze the steamline break event outside containment for a range of conditions.

6.5.2.2 Description of Analyses

The description of the analysis and methods pertaining to the main steamline break mass and energy releases outside containment are presented below.

To determine the effects of plant power level and break area on the mass and energy releases from a ruptured steamline, spectra of both variables have been evaluated as part of the methodology development program documented in Reference 1. At plant power levels of 102 percent and 70 percent, various break sizes have been defined ranging from 0.1 ft^2 to the full double-ended rupture of a main steamline.

A full break spectrum at both power levels (102 percent and 70 percent) has been analyzed at the conditions associated with the Westinghouse-design Model Δ75 steam generator replacement and uprated power. Other assumptions regarding important plant conditions and features are discussed in the following paragraphs.

6.5.2.2.1 Initial Power Level

The initial power that is assumed for steamline break analyses outside containment affects the mass and energy releases and steam generator tube bundle uncovery in two ways. First, the steam generator mass inventory increases with decreasing power levels; this will tend to delay uncovery of the steam generator tube bundle, although the increased steam pressure associated with lower power levels will cause a faster blowdown at the beginning of the transient. Second, the amount of stored energy and decay heat, as well as feedwater temperature, are less for lower power levels; this will result in lower primary temperatures and less primary-to-secondary heat transfer during the steamline break event.

The following power levels were used in the analysis:

- Full power - maximum allowable NSSS power plus uncertainty, i.e., 102 percent of rated power; and

- Near full-power - 70 percent of maximum allowable NSSS power.

For this steam generator replacement and power uprate analysis, the power levels and steamline break sizes are noted in subsection 6.5.2.2.16 of this report.

In general, the plant initial conditions are assumed to be at the nominal value corresponding to the initial power for that case, with appropriate uncertainties included. Tables 6.5.2-1 and 6.5.2-2 identify the values assumed for RCS pressure, RCS vessel average temperature, RCS flow, pressurizer water volume, steam generator water level, and feedwater enthalpy corresponding to each power level analyzed. Steamline break mass releases and superheated steam enthalpies assuming an RCS average temperature at the high end of the T_{avg} window are conservative with respect to similar releases at the low end of the T_{avg} window. At the high end, there is a larger value for the superheated steam enthalpy available for release outside containment. The thermal design flowrate has been used for the RCS flow input consistent with the assumptions documented in Reference 2. The thermal design flowrate is also consistent with other MSLB analysis assumptions related to nonstatistical treatment of uncertainties, as well as RCS thermal-hydraulic inputs related to pressure drops and rod drop time.

Uncertainties on the initial conditions assumed in the analysis for the SGR/Uprating program have been applied only to the RCS average temperature (6°F), the steam generator mass (8 percent narrow-range span), and the power fraction (2 percent) and feedwater enthalpy (2°F) at full power. Nominal values are adequate for the initial conditions associated with pressurizer pressure and pressurizer water level. Uncertainty conditions are only applied to those parameters that could increase the enthalpy of superheated steam discharged out of the break.

6.5.2.2.2 Single-Failure Assumption

The limiting single failure is the failure of one train of safety functions resulting in minimum auxiliary feedwater (AFW) flow and minimum safety injection capability. Variations in AFW flow can affect steamline break mass and energy releases in a number of ways including break mass flowrate, RCS temperature, tube bundle uncovery time and steam superheating. The failure in the AFW system results in a minimum AFW flow to the steam generators; the minimum AFW flow used in the analysis is conservatively based on only one motor-driven AFW pump. The safety injection assumptions are presented in Section 6.5.2.2.11.

6.5.2.2.3 Main Feedwater System

The rapid depressurization that typically occurs following a steamline rupture results in large amounts of water being added to the steam generators through the main feedwater system. However, main feedwater flow has been conservatively modeled by assuming no increase in feedwater flow in response to the increases in steam flow following the steamline break event. This minimizes the total mass addition and associated cooling effects in the steam generators and causes the earliest onset of superheated steam released out of the break.

Isolation of the main feedwater flow is conservatively assumed to be coincident with reactor trip, irrespective of the function that produced the reactor trip signal. This assumption reduces the total mass addition to the steam generators. Closing of the feedwater flow control valves in the main feedwater lines is assumed to be instantaneous with no consideration of associated signal processing or valve stroke time.

Steamline break mass and energy releases assuming a main feedwater temperature at the high end of the feedwater temperature window are conservative with respect to similar releases at the low end of the feedwater temperature window. At the high end, there is more energy available for release outside containment.

6.5.2.2.4 Auxiliary Feedwater System

Generally, within the first few minutes following a steamline break the AFW system (i.e., motor-driven AFW pumps) is initiated on any one of several protection system signals. Addition of auxiliary feedwater to the steam generators will increase the secondary mass available to cover the tube bundle and reduces the amount of superheated steam produced. For this reason, AFW flow is minimized while actuation delays are maximized to accentuate the depletion of the initial secondary-side inventory. The volume of the AFW piping is maximized. A purging of the AFW piping is assumed, since maximum volume delays the injection of colder AFW into the steam generator, following any hotter feedwater resident in the piping up to the isolation valve closest to the steam generator. The less dense resident auxiliary feedwater exhibits a decreased mass addition to the faulted-loop steam generator than if the AFW is introduced directly into the steam generator. The large volume also delays the introduction of colder AFW into any steam generator, which reduces the amount of the cooldown effect on the primary side of the RCS.

6.5.2.2.5 Steam Generator Fluid Mass

A minimum initial steam generator mass in all the steam generators has been used in all of the analyzed cases. The use of a reduced initial steam generator mass minimizes the availability of the heat sink afforded by the steam generators and leads to earlier tube bundle uncovery. The initial mass has been calculated as the value corresponding to the programmed -8 percent narrow-range span level. All steam generator fluid masses are calculated assuming

0 percent tube plugging. This assumption is conservative with respect to the RCS cooldown through the steam generators resulting from the steamline break.

6.5.2.2.6 Steam Generator Reverse Heat Transfer

Once the steamline isolation is complete, the steam generators in the intact loops become sources of energy that can be transferred to the steam generator with the broken steamline. This energy transfer occurs via the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the steam generator tubes could drop below the temperature of the secondary fluid in the intact steam generators, resulting in energy being returned to the primary coolant. This energy is then available to be transferred to the steam generator with the broken steamline. When applicable, the effects of reverse steam generator heat transfer are included in the results.

6.5.2.2.7 Break Flow Model

Piping discharge resistances are not included in the calculation of the releases resulting from the steamline ruptures [Moody Curve for an f(l / D) = 0 is used].

6.5.2.2.8 Steamline Volume Blowdown

There is no contribution to the mass and energy releases from the steam in the secondary plant main steam loop piping and header because the initial volume is saturated steam. With the focus of the MSLB analysis outside containment on maximizing the superheated steam enthalpy, it is presumed that the saturated steam in the loop piping and the header has no adverse effects on the results. The blowdown of the steam in this volume serves to delay the time of tube uncovery in the steam generators and is conservatively ignored. Additional information on the effect of main steam isolation on the superheated steam blowdown is discussed in subsection 6.5.2.2.9.

6.5.2.2.9 Main Steamline Isolation

Steamline isolation is assumed to terminate the blowdown from the intact-loop steam generators; the faulted-loop steam generator is assumed to be unisolable, an indication that the steamline break is upstream of the MSIV. If the MSLB is postulated downstream of the MSIV, the analysis results conservatively account for the continued blowdown from the faulted-loop steam generator as well as minimum AFW flow as discussed in subsection 6.5.2.2.4. However, there is no single failure that would permit more than one steam generator from blowing down through the pipe break.

The crediting of MSIV closure to trap superheated steam is not relevant to this analysis. MSIV closure functions only to isolate the other steam generators from the break location.

The main steamline isolation function is accomplished via the main steam isolation valve in each of the three steamlines. The actuation signal to isolate the main steamlines is received from a

dynamically compensated low steamline pressure setpoint. A delay time of 7 seconds, accounting for delays associated with signal processing plus MSIV stroke time, with unrestricted steam flow through the valve during the valve stroke, has been assumed.

6.5.2.2.10 Protection System Actuations

The protection systems available to mitigate the effects of a MSLB accident outside containment include reactor trip, safety injection, steamline isolation, and auxiliary feedwater. The protection system actuation signals and associated setpoints that have been modeled in the analysis are identified in Table 6.5.2-3. The setpoints used are conservative values with respect to the plant-specific values delineated in the HNP Technical Specifications.

At 102 percent power for break sizes 0.7 ft^2 and larger, the first protection system signal actuated is Low Steamline Pressure (lead/lag compensated in each channel) in the faulted loop. The Low Steamline Pressure signal initiates steamline isolation and safety injection; the safety injection signal produces a reactor trip signal. Main feedwater flow is conservatively assumed to be isolated at the time of reactor trip; motor-driven AFW initiation occurs as a result of the safety injection signal. However, isolation of the AFW flow to the faulted-loop steam generator occurs prior to initiation of the AFW pumps. Isolation of AFW occurs following a steamline ΔP signal. For intermediate-size breaks, from 0.6 ft^2 to 0.4 ft^2, reactor trip is actuated following the Overpower ΔT signal; for break sizes smaller than 0.4 ft^2, reactor trip is actuated following the Low-Low Steam Generator Water Level signal. Main feedwater flow is conservatively assumed to be isolated at the time of reactor trip. Safety injection is started as a result of a Low Pressurizer Pressure signal; steamline isolation occurs later due to Low Steamline Pressure. For the smallest break size analyzed, 0.1 ft^2, the Low Pressurizer Pressure setpoint is not reached; safety injection and steamline isolation occur as a result of Low Steamline Pressure. Auxiliary feedwater flow is initiated following the Low-Low Steam Generator Water Level signal.

At 70 percent power for break sizes 0.8 ft^2 and larger, the first protection system signal actuated is Low Steamline Pressure (lead/lag compensated in each channel) in the faulted loop. The Low Steamline Pressure signal initiates steamline isolation and safety injection; the safety injection signal produces a reactor trip signal. Main feedwater flow is conservatively assumed to be isolated at the time of reactor trip; motor-driven AFW initiation occurs as a result of the safety injection signal. However, isolation of the AFW flow to the faulted-loop steam generator occurs prior to initiation of the AFW pumps. Isolation of AFW occurs following a steamline ΔP signal. For break sizes smaller than 0.8 ft^2, reactor trip and motor-driven AFW initiation are actuated following a Low-Low Steam Generator Water Level signal. Main feedwater flow is conservatively assumed to be isolated at the time of reactor trip. Safety injection is started as a result of a Low Pressurizer Pressure signal; steamline isolation occurs later due to Low Steamline Pressure.

The turbine stop valve is assumed to close instantly following the reactor trip signal.

6.5.2.2.11 Safety Injection System

Minimum SIS flowrates corresponding to the failure of one SIS train have been assumed in this analysis. A minimum SI flow is conservative since the reduced boron addition maximizes a return to power resulting from the RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing steam flow out of the break. The delay time to achieve full SI flow is assumed to be 27 seconds for this analysis with offsite power available. A coincident loss of offsite power is not assumed for the analysis of the steamline break outside containment since the mass and energy releases would be reduced due to the loss of forced reactor coolant flow, resulting in less primary-to-secondary heat transfer.

6.5.2.2.12 Reactor Coolant System Metal Heat Capacity

As the primary side of the plant cools, the temperature of the reactor coolant drops below the temperature of the reactor coolant piping, the reactor vessel, the reactor coolant pumps, and the steam generator thick-metal mass and tubing. As this occurs, the heat stored in the metal is available to be transferred to the steam generator with the broken line. Stored metal heat, however, does not have a major impact on the calculated mass and energy releases. The effects of this RCS metal heat are included in the results using conservative thick-metal masses and heat transfer coefficients.

6.5.2.2.13 Core Decay Heat

Core decay heat generation assumed in calculating the steamline break mass and energy releases is based on the 1979 ANS Decay Heat + 2σ model (Reference 3). This version of the decay heat input has been applied previously to other HNP licensing-basis safety analyses (FSAR Chapter 15, Reference 15.0.10-3).

6.5.2.2.14 Rod Control

The rod control system is conservatively assumed to be in manual operation for all steamline break analyses.

6.5.2.2.15 Core Reactivity Coefficients

Conservative core reactivity coefficients corresponding to end-of-cycle conditions are used to maximize the reactivity feedback effects resulting from the steamline break. Use of maximum reactivity feedback results in higher power generation if the reactor returns to criticality, thus maximizing heat transfer to the secondary side of the steam generators.

6.5.2.2.16 Description of Analysis Cases

The system transient that provides the break flows and enthalpies of the steam release through the steamline break outside containment has been analyzed with the LOFTRAN (Reference 4) computer code. Blowdown mass and energy releases determined using LOFTRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant system thick-metal heat storage, and reverse steam generator heat transfer. The use of the LOFTRAN code for the analysis of the MSLB with superheated steam M&E releases is documented in Supplement 1 of WCAP-8822 (Reference 2), which has been reviewed and approved by the NRC for use in analyzing main steamline breaks, and in Reference 1 for MSLBs outside containment. The LOFTRAN code has been utilized previously for the HNP licensing-basis safety analyses.

The HNP NSSS has been analyzed to determine the transient mass releases and associated superheated steam enthalpy values outside containment following a steamline break event. The tables of mass flowrates and steam enthalpies are used as input conditions to the environmental evaluation of safety-related electrical equipment in the main steam tunnel.

The following licensing-basis cases of the MSLB outside containment have been analyzed at the noted conditions for the SGR/Uprating program.

- At 102 percent power, break sizes of 4.2, 2.0, 1.4, 1.2, 1.0, 0.9, 0.8, 0.7, 0.6, 0.5, 0.4, 0.3, 0.2, and 0.1 ft^2

- At 70 percent power, break sizes of 4.2, 2.0, 1.4, 1.2, 1.0, 0.9, 0.8, 0.7, 0.6, 0.5, 0.4, 0.3, 0.2, and 0.1 ft^2

Each MSLB outside containment is represented as a nonmechanistic split rupture (crack area). The largest break is postulated as a crack area equivalent to a single-ended pipe rupture. The actual cross-sectional flow area of the steamline outside containment is 4.94 ft^2, and the flow area of the steam header piping is 10.66 ft^2. However, since the break flowrate is limited by the total cross-sectional flow area of the three integral flow restrictors, the maximum break size is limited to 4.2 ft^2 rather than the actual pipe break area. Prior to steamline isolation, the break area is represented by the spectrum noted above. After steamline isolation, the break area is limited by the smaller of the integral steam generator flow restrictor (1.4 ft^2) or the defined break size.

6.5.2.3 Acceptance Criteria

The main steamline break is classified as an ANS Condition IV event, an infrequent fault. The acceptance criteria associated with the steamline break event resulting in a mass and energy release outside containment is based on an analysis which provides sufficient conservatism to ensure that the equipment qualification temperature envelope is maintained. The specific criteria

applicable to this analysis are related to the assumptions regarding power level, stored energy, the break flow model, steamline and feedwater isolation, and main and auxiliary feedwater flow such that superheated steam resulting from tube bundle uncovery in the steam generators is accounted for and maximized. These analysis assumptions have been included in this steamline break mass and energy release analysis as discussed in subsection 6.5.2.2 of this report. The tables of mass flowrates and steam enthalpy values for each of the steamline break cases noted in the previous section are used as input to the environmental evaluation of safety-related electrical equipment in the main steam tunnel.

6.5.2.4 Results

Using the MSLB analysis methodology documented in Reference 1 as a basis, including parameter changes associated with the SGR/Uprating, the mass and energy release rates for each of the steamline break cases noted in subsection 6.5.2.2.16 have been developed for use in the environmental evaluation of safety-related electrical equipment in the main steam tunnel. Tables 6.5.2-4 and 6.5.2-5 provide the sequence of events for the various steamline break sizes at 102 percent and 70 percent power, respectively.

6.5.2.5 Conclusions

The mass releases and associated steam enthalpy values from the spectrum of steamline break cases outside containment have been analyzed at the conditions defined by the SGR/Uprating program. The analysis methods delineated in subsection 6.5.2.2 have been included in the steamline break analysis such that conservative mass and energy releases are calculated. The 102 percent power and 70 percent power MSLB M&E releases outside containment obtained with the Model Δ75 replacement steam generators at the uprated power of 2912.4 MWt were generated using a method consistent with MSLB M&E releases that would be obtained with the Model Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

These main steamline break mass and energy releases have been calculated on a HNP plant-specific basis to address the superheated steam issue identified in the NRC Information Notice 84-90 (Main Steam Line Break Effect on Environmental Qualification of Equipment, December 7, 1984). The mass releases and associated steam enthalpy values discussed in this section have been provided for use in the environmental evaluation of safety-related electrical equipment in the main steam tunnel in support of the HNP SGR/Uprating program. The results of the steam tunnel analysis and environmental evaluation for the SGR/Uprating are provided in the BOP Licensing Report.

6.5.2.6 References

1. WCAP-10961, Rev. 1, (Proprietary), "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment, Report to the Westinghouse Owners Group High Energy Line Break/Superheated Blowdowns Outside Containment Subgroup," October 1985.

2. WCAP-8822 (Proprietary) and WCAP-8860 (Nonproprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-S1-A (Nonproprietary), "Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986; WCAP-8822-S2-P-A (Proprietary) and WCAP-8860-S2-A (Nonproprietary), "Supplement 2 – Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs," September 1986.

3. ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.

4. WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), "LOFTRAN Code Description," April 1984.

Table 6.5.2-1 Nominal Plant Parameters for SGR/Uprating* (MSLB M&E Releases Outside Containment)	
Nominal Conditions	
NSSS Power, MWt	2912.4
Core Power, MWt	2900
Reactor Coolant Pump Heat, MWt	12.4
Reactor Coolant Flow (total), gpm (Thermal Design Flow)	277,800
Pressurizer Pressure, psia	2250
Core Bypass, %	7.1
Reactor Coolant Vessel Average Temperature, °F	588.8*
Steam Generator**	
Steam Temperature, °F	545.9
Steam Pressure, psia	1011
Steam Flow, 10^6 lbm/hr (Plant Total)	12.87
Feedwater Temperature, °F	440
Zero-Load Temperature, °F	557

Notes:

* Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T_{avg} window.

** Steam generator performance data used in the analysis is conservatively high for steam temperature and pressure. This data also corresponds to best-estimate RCS flow conditions.

Table 6.5.2-2 Initial Condition Assumptions for SGR/Uprating* (MSLB M&E Releases Outside Containment)		
Initial Conditions	**Power Level (%)**	
Parameter	**102**	**70**
RCS Average Temperature (°F)	594.8*	585.3*
RCS Flowrate (gpm) (Thermal Design Flow)	277,800	277,800
RCS Pressure (psia)	2250	2250
Pressurizer Water Volume (ft^3)	836.5	704.9
Feedwater Enthalpy (Btu/lbm)	421.7	371.7
SG Pressure (psia)**	1059	1082
SG Water Level, faulted/intact (% NRS)	48.7	48.7

Notes:

* Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T_{avg} window; temperatures include applicable calorimetric uncertainties.

** The noted SG pressures are determined at the steady-state conditions defined by the RCS average temperatures, including applicable uncertainties except at 0% power.

Table 6.5.2-3 **Protection System Actuation Signals and Safety System Setpoints for SGR/Uprating** **(MSLB M&E Releases Outside Containment)**
<u>Reactor Trip</u> 2/3 Low-Low Steam Generator Water Level in any loop – 0% narrow-range span 2/3 Low Pressurizer Pressure - 1935 psia 2/4 Power-Range High Neutron Flux – 118% rated thermal power 2/3 Overtemperature ΔT K1 = 1.32 K2 = 0.0224 K3 = 0.0012 dynamic compensation lead - 22 seconds lag - 4 seconds 2/3 Overpower ΔT K4 = 1.18 K5 = 0.0 K6 = 0.0 dynamic compensation rate lag - 13 seconds Safety Injection
<u>Safety Injection</u> 2/3 Low Pressurizer Pressure – 1714.3 psia 2/3 Low Steamline Pressure in any loop – 556.9 psia dynamic compensation lead - 50 seconds lag - 5 seconds
<u>Steamline Isolation</u> 2/3 Low Steamline Pressure in any loop – 556.9 psia dynamic compensation lead - 50 seconds lag - 5 seconds
<u>Feedwater Isolation</u> Coincident with Reactor Trip (conservative assumption)
<u>Auxiliary Feedwater Initiation (motor-driven AFW pumps)</u> 2/3 Low-Low Steam Generator Water Level in any loop – 0% narrow-range span Safety Injection

Table 6.5.2-4

Transient Summary for the Spectrum of Breaks at 102% Power - Outside Containment

Power Level (% Nom)	Break Size (ft^2)	Reactor Trip Signal	Time of Rod Motion (sec)	Feedwater Isolation (sec)	Safety Injection Signal	Time Safety Injection Starts (sec)	Steamline Isolation (sec)	Time of AFW Actuation (sec)	Time of AFW Isolation (sec)	SG Tube Uncovery in Faulted SG (sec)
102	4.2	LSP	2.9	2.9	LSP	27.9	7.9	62.4	8.5	45.5
102	2.0	LSP	3.2	3.2	LSP	28.2	8.2	62.7	8.5	47.0
102	1.4	LSP	3.8	3.8	LSP	28.8	8.8	63.3	9.2	48.5
102	1.2	LSP	4.3	4.3	LSP	29.3	9.3	63.8	9.7	53.0
102	1.0	LSP	5.1	5.1	LSP	30.1	10.1	64.6	10.5	59.0
102	0.9	LSP	5.7	5.7	LSP	30.7	10.7	65.2	11.2	63.0
102	0.8	LSP	6.8	6.8	LSP	31.8	11.8	66.3	12.3	68.5
102	0.7	LSP	10.3	10.3	LSP	35.3	15.3	69.8	15.5	76.0
102	0.6	OPΔT	28.0	28.0	LPP	107.9	151.8	115.8	153.0	167.5
102	0.5	OPΔT	33.6	33.6	LPP	128.0	272.8	125.5	273.9	185.0
102	0.4	OPΔT	48.6	48.6	LPP	165.6	338.9	141.4	340.1	211.5
102	0.3	LSGWL	119.9	119.9	LPP	274.6	404.8	177.9	406.4	226.3
102	0.2	LSGWL	173.8	173.8	LPP	419.3	599.2	231.8	603.3	304.0
102	0.1	LSGWL	337.4	337.4	LSP	1136.6	1116.6	395.4	1122.8	520.5

LSP- Low Steamline Pressure

LSGWL - Low-Low Steam Generator Water Level

LPP - Low Pressurizer Pressure

OPΔT - Overpower ΔT

Table 6.5.2-5
Harris Nuclear Plant
Transient Summary for the Spectrum of Breaks at 70% Power - Outside Containment

Power Level (% Nom)	Break Size (ft^2)	Reactor Trip Signal	Time of Rod Motion (sec)	Feedwater Isolation (sec)	Safety Injection Signal	Time Safety Injection Starts (sec)	Steamline Isolation (sec)	Time of AFW Actuation (sec)	Time of AFW Isolation (sec)	SG Tube Uncovery in Faulted SG (sec)
70	4.2	LSP	2.7	2.7	LSP	27.7	7.7	62.2	8.4	57.0
70	2.0	LSP	3.3	3.3	LSP	28.3	8.3	62.8	8.7	60.0
70	1.4	LSP	4.0	4.0	LSP	29.0	9.0	63.5	9.5	61.0
70	1.2	LSP	4.5	4.5	LSP	29.5	9.5	64.0	10.0	66.5
70	1.0	LSP	5.4	5.4	LSP	30.4	10.4	64.9	10.9	74.5
70	0.9	LSP	6.1	6.1	LSP	31.1	11.1	65.6	11.7	79.5
70	0.8	LSP	7.5	7.5	LSP	32.5	12.5	67.0	13.1	87.0
70	0.7	LSGWL	54.6	54.6	LPP	124.3	117.5	112.6	118.4	152.0
70	0.6	LSGWL	62.3	62.3	LPP	141.7	157.5	120.3	157.4	188.0
70	0.5	LSGWL	72.9	72.9	LPP	166.4	317.3	130.9	316.4	212.0
70	0.4	LSGWL	88.8	88.8	LPP	203.2	398.4	146.8	397.2	254.0
70	0.3	LSGWL	114.9	114.9	LPP	265.9	520.9	172.9	523.3	314.0
70	0.2	LSGWL	166.7	166.7	LPP	399.1	758.6	224.7	762.4	380.3
70	0.1	LSGWL	320.9	320.9	LPP	890.5	1497.8	378.9	1515.0	566.8

LSP - Low Steamline Pressure
LSGWL - Low-Low Steam Generator Water Level
LPP - Low Pressurizer Pressure

6.5.3 Steam Releases for Radiological Dose Analysis

6.5.3.1 Introduction

Vented steam releases have been calculated for the Loss-of-Offsite AC Power, Locked Rotor, and Steamline Break events to support the SGR/Uprating program. Information documented in Tables 15.1.5-5 and 15.2.6-5 of the HNP FSAR includes steam releases and feedwater flows. These steam releases and feedwater flows are used as input to the radiological dose analysis that is required to support the SGR/Uprating program.

6.5.3.2 Description of Analyses

The description of the analysis and methods pertaining to the calculated steam releases and feedwater flows for use in the radiological analysis are presented below.

Steam dump will be required until the reactor can be placed on the residual heat removal (RHR) system. It has been confirmed that eight hours of steam release will occur prior to placing the plant in the RHR mode of operation. In the event of the loss of one train of electric power, the power supply to one of the two RHR loop suction valves in an RHR train must be rewired to the functioning train of electric power. This problem is assumed to be identified and remedied within the 8-hour time frame of the steam releases from the steam generator PORVs. After the first 2 hours, it is assumed the plant will have cooled down and stabilized at no-load conditions. The additional 6 hours are required to cool down and depressurize from no-load conditions to the RHR operating conditions.

The Steam Generator Blowdown System was assumed to be isolated for the events analyzed. This is conservative with regard to the calculation of the steam released to the atmosphere.

The amount of steam released to the atmosphere depends on the sensible heat and decay heat generated while reducing the temperature from the full-power value to the shutdown conditions. A calculation is performed to determine the amount of steam that is dissipated through the atmospheric steam release.

The total RCS energy at the end of the first 2-hour interval is subtracted from the sum of the initial RCS energy and the decay heat generated during this interval. For the Steamline Break event, it is conservative to assume that the contents of the faulted-loop steam generator blow down within the first 2 hours with no energy extraction from the RCS (i.e., no temperature decrease) due to the blowdown. Likewise, the total RCS energy at the end of the 2-to-8-hour interval is subtracted from the sum of the RCS energy and the decay heat generated during this 6-hour interval.

An energy balance during both of these intervals is used to calculate the mass of auxiliary feedwater injected during the cooldown interval. The mass of feedwater injected is used to

calculate the steam mass vented to the environment through the intact-loop steam generators. For the Loss-of-Offsite AC Power and Locked Rotor events, three intact loops have been used in the steam release calculations; for the Steamline Break event, two intact loops have been used for this calculation. An additional calculation is performed for the Steamline Break event in which the contents of the faulted-loop steam generator blow down during the first 2-hour interval.

6.5.3.3 Acceptance Criteria

There are no specific acceptance criteria associated with the calculation of the steam releases and feedwater flows used as input to the radiological dose analyses. Tables of steam releases and feedwater flows for each of the cooldown intervals for each of these three transients are used as input to the radiological dose analysis in support of the HNP SGR/Uprating.

6.5.3.4 Results

Table 6.5.3-1 summarizes the vented steam releases from the intact-loop steam generators as well as feedwater flows for the 0–2 hour time period and the 2–8 hour time period for the Loss-of-Offsite AC Power, Locked Rotor, and Steamline Break events. These two time periods are documented to support the SGR/Uprating program.

For the Steamline Break event, additional steam is released through the faulted-loop steam generator from the initiation of the transient up through the time at which main and auxiliary feedwater flows are assumed to be terminated. The additional steam comes from two components: the initial mass of steam in the steam generator and feedwater addition subsequent to the initiation of the Main Steamline Break. The steam mass from the faulted-loop steam generator is 162,000 lbm. This steam generator mass is not used in the energy balance which predicts the results listed in Table 6.5.3-1. The feedwater component of the steam mass may be treated as clean since it is injected into the faulted steam generator after the initiation of the steamline break and is not subjected to any primary-to-secondary coolant leakage. The feedwater mass to the faulted steam generator is also not used in the energy balance which predicts the results in Table 6.5.3-1.

6.5.3.5 Conclusions

The steam releases and feedwater flows have been calculated at the conditions defined for the SGR/Uprating for the HNP Loss-of-Offsite AC Power, Locked Rotor, and Steamline Break events. The analysis methods delineated in subsection 6.5.3.2 have been included in the steam release calculations for each transient such that the results are consistent with and continue to comply with the current HNP licensing-basis/acceptance requirements. The calculated steam releases and feedwater flows obtained with the Model Δ75 replacement steam generators at the uprated power of 2912.4 MWt bound operation with the Model Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt. The results of the radiological dose analysis are provided in the BOP Licensing Report for the SGR/Uprating program.

Table 6.5.3-1 Vented Steam Released and Feedwater Flows				
Event	**Vented Steam Release**		**Feedwater Flow**	
	0–2 hours	**2–8 hours**	**0–2 hours**	**2–8 hours**
Loss-of-Offsite AC Power and Locked Rotor	364,000 lbm	939,000 lbm	508,000 lbm	1,052,000 lbm
Steamline Break	386,000 lbm	892,000 lbm	482,000 lbm	967,000 lbm

6.6 LOCA Hydraulic Forces

6.6.1 Introduction

The purpose of an analysis of the loss-of-coolant accident (LOCA) hydraulic forces is to generate the hydraulic forcing functions that occur on Reactor Coolant System (RCS) components as a result of a postulated LOCA.

The hydraulic forcing functions that occur as a result of a postulated LOCA are calculated assuming a limiting break location and break area. The limiting break location and area vary with the RCS component under consideration, but historically the limiting postulated breaks are a limited displacement reactor pressure vessel (RPV) inlet/outlet nozzle break or a double-ended guillotine (DEG) reactor coolant pump (RCP)/steam generator (SG) inlet/outlet nozzle break (Reference 1). The NRC's recent revision to General Design Criteria (GDC)-4 allows main coolant piping breaks to be "excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping." This exemption is generally referred to as leak-before-break (LBB). For HNP, the applicability of a leak-before-break design basis was approved in Reference 2. The technical justification for application of LBB to HNP is documented in Reference 3. LBB licensing allows RCS components to be evaluated for LOCA integrity considering the next most limiting auxiliary line breaks.

6.6.2 Description of Analyses and Evaluations

For the LOCA hydraulic forces analysis, the plant parameters considered to be most critical are: pressurizer pressure (2288 psia including uncertainty) and RCS hot leg (T_{hot} = 600.6°F including uncertainty) and cold leg (T_{cold} = 529.8°F including uncertainty) temperatures associated with full-power operation. Other plant parameters that are of importance for the LOCA hydraulic forces are primary side geometry and hydraulic losses in the RCS, and inputs (such as fuel mass and stiffness, and steam generator vertical divider plate mass and natural frequency in air), which affect the flexible walls modeled (core barrel and steam generator vertical divider plate).

The NRC-approved MULTIFLEX computer code (Reference 4) is used to generate the transient hydraulic forcing functions on the reactor vessel and internals due to a postulated rupture in the RCS. Hydraulic forcing functions on the RCS loop piping and steam generators are evaluated using the established LOCA forces sensitivities to changes in RCS temperatures, and the reduced break area associated with LBB licensing. The MULTIFLEX code calculates the thermal-hydraulic transient within the RCS and considers subcooled, transition and two-phase (saturated) blowdown regimes. The code employs the method of characteristics to solve the conservation laws, assuming one dimensional flow and a homogeneous liquid-vapor mixture. The RCS is divided into subregions in which each subregion is regarded as an equivalent pipe. A complex network of these equivalent pipes is used to represent the entire primary RCS.

A coupled fluid-structure interaction is incorporated into the MULTIFLEX code by accounting for the deflection of the constraining boundaries, which are represented by separate spring-mass oscillator systems. For the reactor vessel/internals analysis, the reactor core barrel is modeled as an equivalent beam with the structural properties of the core barrel in a plane parallel to the broken inlet nozzle. Horizontally, the barrel is divided into ten segments, with each segment consisting of three walls. Mass and stiffness matrices that are obtained from an independent modal analysis of the reactor core barrel are applied in the equations of structural vibration at each of the ten mass point locations. Horizontal forces are then calculated by applying the spatial pressure variation to the wall area at each of the elevations representative of the ten mass points of the beam model. The resultant core barrel motion is then translated into an equivalent change in flow area in each downcomer annulus flow channel. At every time increment, the code iterates between the hydraulic and structural subroutines of the program at each location confined by a flexible wall. For the reactor pressure vessel and specific vessel internal components, the MULTIFLEX code generates the LOCA pressure transient that is input to the LATFORC and FORCE2 post-processing codes (Reference 4). These codes, in turn, are used to calculate the actual forces on the various components.

Steam generator hydraulic transient time history data is extracted directly from the MULTIFLEX output, as described in Reference 5.

The LATFORC computer code employs the field pressures generated by MULTIFLEX code, together with geometric vessel information (component radial and axial lengths), to determine the horizontal forces on the vessel wall, core barrel, and thermal shield. The LATFORC code represents the downcomer region with a model that is consistent with the model used in the MULTIFLEX blowdown calculations. The downcomer annulus is subdivided into cylindrical segments, formed by dividing this region into circumferential and axial zones. The results of the MULTIFLEX/LATFORC analysis of the horizontal forces are calculated for the initial 500 msec of the blowdown transient and are stored in a computer file. These forcing functions, combined with vertical LOCA hydraulic forces, seismic, thermal, and system shaking loads, are used by the cognizant structural groups to determine the resultant mechanical loads on the reactor pressure vessel and vessel internals.

The FORCE2 computer code calculates the hydraulic forces that the RCS coolant exerts on the vessel internals in the vertical direction. The FORCE2 code uses a detailed geometric description of the vessel components and the transient pressures, mass velocities, and densities computed by the MULTIFLEX code. The analytical basis for the derivation of the mathematical equations employed in the FORCE2 code is the one-dimensional conservation of linear momentum. Note that the computed vertical forces do not include body forces on the vessel internals, such as deadweight or buoyancy. When the vertical forces on the reactor pressure vessel internals are calculated, pressure differential forces, flow stagnation forces, unrecoverable orifice losses, and friction losses on the individual components are considered. These force components are then summed together, depending upon the significance of each, to yield the total vertical force acting on a given component. The results of the MULTIFLEX/FORCE2 analysis

of the vertical forces are calculated for the initial 500 msec of the blowdown transient and are stored in a computer file. These forcing functions, combined with horizontal LOCA hydraulic forces, seismic, thermal, and system shaking loads, were used in the structural evaluations previously presented to determine the resultant mechanical loads on the vessel and vessel internals.

The loop forces analysis was completed using the THRUST post-processing code. The THRUST code is used to generate the X, Y and Z directional component forces during a LOCA blowdown from the RCS pressure, density, and mass flux. The THRUST code (previously named STHRUST) is described and documented in Reference 6.

With the LBB licensing status of the HNP, only 3 of the original 11 postulated break locations will apply for the loop forces calculations. These are the three branch line breaks: accumulator line, pressurizer surge line, and RHR line. Again, the same three branch line breaks (branch lines nearest the reactor vessel) were postulated for the vessel forces calculation (the accumulator line break from loop 1 or 2, the RHR line break from loop 3, and the pressurizer line break from loop 2). For the steam generator forces calculation, the same branch line breaks were postulated; however, the branch lines nearest the steam generator were used, rather than the branch lines nearest the reactor vessel (the accumulator line break from loop 3, the RHR line break from loop 1, and the pressurizer line break from loop 2).

6.6.3 Acceptance Criteria

There are no specific acceptance criteria on results of LOCA forces analyses by themselves. LOCA Forces are used as input to other component qualification analyses.

6.6.4 Results

6.6.4.1 Reactor Vessel and Vessel Internals

Vessel and vessel internals LOCA hydraulic forcing functions were generated using three postulated auxiliary line breaks. An accumulator line break, a pressurizer surge line break and RHR line break were analyzed using a flexible beam core barrel MULTIFLEX model (for fluid-structure interaction). Vessel LOCA forces were analyzed for the HNP SGR/Uprating program with inputs that reflect the change in fuel product used at HNP from Westinghouse 17x17 Vantage 5 to Siemens Power Corporation 17x17 Fuel. As a result, this impacted the values of input parameters for modeling the core, relative to flow area and hydraulic losses. It also affected the beam model inputs representing the flexible core barrel. The vessel LOCA forces were reanalyzed and fuel/vessel qualification will be based on the new vessel forces.

6.6.4.2 RCS Loop Piping and Steam Generators

As with the vessel forces, the Loop LOCA Forces and the Steam Generator Forces were generated using three postulated auxiliary line breaks (Accumulator line, pressurize line and RHR line break). Loop hydraulic forces and Steam Generator forces had to be reanalyzed due to specific loop branch line locations and the changes in the RCS conditions of temperature and pressurizer pressure and uncertainties.

6.6.5 Conclusions

Vessel LOCA Forces were analyzed for the HNP SGR/Uprating program with inputs that reflect the change in fuel product used at HNP from Westinghouse 17x17 Vantage 5 to Siemens Power Corporation 17x17 fuel. Loop LOCA Forces and Steam Generator Forces were reanalyzed due to specific loop branch line locations and HNP specific conditions of temperature and pressure.

In all cases, the basic magnitude of the LOCA hydraulic forces remained close to the values currently calculated with differences in specific components where input values have changed from the existing analyses (such as fuel type, branch line location and HNP replacement SG design). The results of the analysis, namely vessel, loop and steam generator forces at the SGR/Uprating conditions bound operation with the Δ75 replacement steam generators at the current NSSS power level of 2787.4 MWt. The results were used by the cognizant component analysis groups for use in the project. The acceptability of the LOCA hydraulic forces is demonstrated in the structural analyses for the components as described in Sections 5.2, 5.5 and 5.7.

6.6.6 References

1. WCAP-8082-P-A, "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop," approved by R. Salvatori, January 1975.

2. NRC Docket 50-400, "Request for Exemption From a Portion of General Design Criteria 4 of Appendix A to 10 CFR Part 50 Regarding the Need to Analyze Large Primary Loop Pipe Ruptures as a Structural Design Basis for Shearon Harris Nuclear Power Plant, Unit 1," G. W. Knighton, June 5, 1985.

3. WCAP-14549, "Technical Justification for Eliminating Large Loop Pipe Rupture as the Structural Design Basis for the Shearon Harris Unit 1 Nuclear Power Plant," December 1996.

4. WCAP-8708-P-A, "MULTIFLEX, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," K. Takeuchi, et al., September 1977.

5. WCAP-7832-A, "Evaluation of Steam Generator Tube, Tube Sheet, and Divider Plate Under Combined LOCA Plus SSE Conditions," P. De Rosa, April 1978.

6. WCAP-8252, Rev. 1, "Documentation of Selected Westinghouse Structural Analysis Computer Codes," K.M. Vashi, May 1977.

6.7 Reactor Trip System/Engineered Safety Feature Actuation System Setpoints

6.7.1 Introduction

The Technical Specification Reactor Trip System (RTS)/Engineered Safety Feature Actuation System (ESFAS) setpoints have been reviewed for Harris Nuclear Plant (HNP) operation at SGR/Uprating conditions. As part of the review, Technical Specification changes are recommended consistent with the Westinghouse setpoint methodology.

The Technical Specification Allowable Values are used to determine operability of the process instrumentation, and have been changed to reflect the calibration tolerances used in HNP plant calibration procedures. HNP calibration procedures use two-sided calibration tolerances for calibration of the instrumentation. An Allowable Value must be linked to the Technical Specification Trip Setpoint to determine operability. The Allowable Value is determined by adding the calibration tolerance to the Trip Setpoint (for a high trip setpoint), or by subtracting the calibration tolerance from the Trip Setpoint (for a low trip setpoint).

6.7.2 Description of Analyses and Evaluations

The uncertainty analysis uses the "square root of the sum of the squares" to combine the uncertainty components of an instrument channel in an appropriate combination of those components, or groups of components, which are statistically independent. Those uncertainties that are not independent are arithmetically summed to produce groups that are independent of each other, which can then be statistically combined.

6.7.3 Acceptance Criteria

The acceptance criteria for the RTS/ESFAS setpoints is: Margin ≥ 0.

Margin is defined as the difference between the Total Allowance (TA) and the Channel Statistical Allowance (CSA). Total Allowance is the difference between the limiting FSAR Chapter 15 safety analysis limit and the Technical Specification Trip Setpoint (in percent of instrument span). Channel Statistical Allowance is the statistical combination of the instrument channel uncertainty components (in percent of instrument span).

6.7.4 Results

Tables 6.7-1 and 6.7-2 list both the current and power uprate RTS/ESFAS setpoint values for each impacted function and parameter. Incorporating these Technical Specification changes will ensure that HNP will operate in a manner consistent with the FSAR assumptions. The results of the RTS/ESFAS setpoint analysis are provided in Reference 1.

6.7.5 Conclusions

The results obtained with the Δ75 replacement steam generators at the uprated NSSS power of 2912.4 MWt bound operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt.

6.7.6 References

1. WCAP-15249, Rev. 0, "Westinghouse Protection System Setpoint Methodology for Harris Nuclear Plant (For Uprate to 2912.4 MWt - NSSS Power and Replacement Steam Generators)," January 2000.

Table 6.7-1 Summary of the Technical Specification Reactor Trip System Setpoint Changes				
	Trip Setpoint		**Allowable Value**	
	Current Value	**SGR/Uprating Value**	**Current Value**	**SGR/Uprating Value**
Power Range, Neutron Flux – High, Reactor Trip Functional Unit 2				
Low Setpoint	≤25.0% RTP	25.0% RTP	≤27.1% RTP	≤25.6% RTP
High Setpoint	≤109.0% RTP	109.0% RTP	≤111.1% RTP	≤109.6% RTP
Power Range, Neutron Flux, Positive Rate – High, Reactor Trip Functional Unit 3				
High Setpoint	≤5.0% RTP with time constant ≥2 seconds	5.0% RTP with time constant ≥2 seconds	≤6.3% RTP with time constant ≥2 seconds	≤5.6% RTP with time constant ≥2 seconds
Power Range, Neutron Flux, Negative Rate – High, Reactor Trip Functional Unit 4				
High Setpoint	≤5.0% RTP with time constant ≥2 seconds	5.0% RTP with time constant ≥2 seconds	≤6.3% RTP with time constant ≥2 seconds	≤5.6% RTP with time constant ≥2 seconds
Intermediate Range, Neutron Flux – High, Reactor Trip Functional Unit 5				
High Setpoint	≤25.0% RTP	25.0% RTP	≤30.9% RTP	≤27.4% RTP
Source Range, Neutron Flux – High, Reactor Trip Functional Unit 6				
High Setpoint	$\leq 1.0 \times 10^5$ cps	1.0×10^5 cps	$\leq 1.4 \times 10^5$ cps	$\leq 1.05 \times 10^5$ cps

Table 6.7-1 (Cont.) Summary of the Technical Specification Reactor Trip System Setpoint Changes				
	Trip Setpoint		Allowable Value	
	Current Value	SGR/Uprating Value	Current Value	SGR/Uprating Value
Overtemperature ΔT Reactor Trip Functional Unit 7				
K_1	≤1.17	1.185	≤2.1% ΔT	≤0.35% ΔT span for ΔT ≤0.52% ΔT span for T_{avg} ≤0.06% ΔT span for Pressure ≤0.12% ΔT span for ΔI
K_2	0.0224/°F	0.0224/°F	N/A	N/A
K_3	0.001072/psig	0.0012/psig	N/A	N/A
T′	Nominal T_{avg} ≤580.8°F	Reference T_{avg} ≤588.8°F	N/A	N/A
-ΔI Gain	2.36	1.75	N/A	N/A
+ΔI Gain	1.57	1.50	N/A	N/A
f(ΔI) Penalty Dead-band	-21.6%, to +12.0%	-21.6%, to +12.0%	N/A	N/A
τ_1	8 seconds	0 seconds	N/A	N/A
τ_2	3 seconds	0 seconds	N/A	N/A
τ_3	0 seconds	4 seconds	N/A	N/A
τ_4	20 seconds	22 seconds	N/A	N/A
τ_5	4 seconds	4 seconds	N/A	N/A
τ_6	0 seconds	0 seconds	N/A	N/A

Table 6.7-1 (Cont.) Summary of the Technical Specification Reactor Trip System Setpoint Changes				
	Trip Setpoint		**Allowable Value**	
	Current Value	**SGR/Uprating Value**	**Current Value**	**SGR/Uprating Value**
Overpower ΔT Reactor Trip Functional Unit 8				
K_4	≤1.079	1.12	≤2.3%ΔT	≤0.35%ΔT span for ΔT ≤0.05%ΔT span for T_{avg}
K_5	0.02/°F for increasing T, 0 for decreasing T	0.02/°F for increasing T, 0 for decreasing T	N/A	N/A
K_6	0.002/°F for T>T"; 0 for T≤T"	0.002/°F for T>T"; 0 for T≤T"	N/A	N/A
T″	Indicated T_{avg} ≤580.8°F	Reference T_{avg} ≤588.8°F	N/A	N/A
τ_1	8 seconds	0 seconds	N/A	N/A
τ_2	3 seconds	0 seconds	N/A	N/A
τ_3	0 seconds	4 seconds	N/A	N/A
τ_6	0 seconds	0 seconds	N/A	N/A
τ_7	10 seconds	13 seconds	N/A	N/A

Table 6.7-1 (Cont.) Summary of the Technical Specification Reactor Trip System Setpoint Changes				
	Trip Setpoint		Allowable Value	
	Current Value	SGR/Uprating Value	Current Value	SGR/Uprating Value
Pressurizer Pressure – Low, Reactor Trip Functional Unit 9				
Low Setpoint	≥1960 psig	1960 psig	≥1946 psig	≥1956 psig
Pressurizer Pressure – High, Reactor Trip Functional Unit 10				
High Setpoint	≤2385 psig	2385 psig	≤2399 psig	≤2389 psig
Pressurizer Water Level – High, Reactor Trip Functional Unit 11				
High Setpoint	≤92% of span	92% of span	≤93.8% of span	≤92.5% of span
Reactor Coolant Flow – Low, Reactor Trip Functional Unit 12				
Low Setpoint	≥90.5% loop indicated flow[(a)]	90.5% loop indicated flow[(a)]	≥89.5% loop indicated flow[(a)]	≥90.1% loop indicated flow[(a)]
Steam Generator Water Level – Low-Low, Reactor Trip Functional Unit 13				
Low-Low Setpoint	≥38.5% of span	25.0% of span	≥36.5% of span	≥24.5% of span
Steam Generator Water Level – Low, Reactor Trip Functional Unit 14				
Low Setpoint	≥38.5% of span	25.0% of span	≥36.5% of span	≥24.5% of span
Coincident With Steam/Feedwater Flow Mismatch – High Functional Unit 14				
High Setpoint	≤40.0% of full steam flow at RTP	40.0% of full steam flow at RTP	≤43.1% of full steam flow at RTP	≤40.6% of full steam flow at RTP

Table 6.7-1 (Cont.) Summary of the Technical Specification Reactor Trip System Setpoint Changes				
	Trip Setpoint		**Allowable Value**	
	Current Value	**SGR/Uprating Value**	**Current Value**	**SGR/Uprating Value**
Undervoltage – RCPs, Reactor Trip Functional Unit 15				
Low Setpoint	≥5148 volts	5148 volts	≥4920 volts	≥5053 volts
Underfrequency – RCPs, Reactor Trip Functional Unit 16				
Low Setpoint	≥57.5 Hz	57.5 Hz	≥57.3 Hz	≥57.4 Hz
Turbine Trip - Low Fluid Oil Pressure Functional Unit 17.a				
Low Setpoint	≥1000 psig	1000 psig	≥950 psig	≥972 psig
Turbine Trip - Turbine Throttle Valve Closure Functional Unit 17.b				
Low Setpoint	≥1% open	4.76% open	≥1% open	≥3.17% open
Reactor Trip System Interlocks Functional Unit 19.a				
Intermediate Range Neutron Flux, P-6	$\geq 1 \times 10^{-10}$ amp	1×10^{-10} amp	$\geq 6 \times 10^{-11}$ amp	$\geq 9 \times 10^{-11}$ amp
Functional Unit 19.b.1				
Low Power Reactor Trips Block, P-7 (P-10 Input)	≤10.0% RTP	10.0% RTP	≤12.1% RTP	≤10.6% RTP
Functional Unit 19.b.2				
Low Power Reactor Trips Block, P-7 (P-13 Input)	≤10.0% RTP Turbine Impulse Pressure Equivalent	10.0% RTP Turbine Impulse Pressure Equivalent	≤12.1% RTP Turbine Impulse Pressure Equivalent	≤10.6% RTP Turbine Impulse Pressure Equivalent

Table 6.7-1 (Cont.) Summary of the Technical Specification Reactor Trip System Setpoint Changes				
	Trip Setpoint		**Allowable Value**	
	Current Value	**SGR/Uprating Value**	**Current Value**	**SGR/Uprating Value**
Reactor Trip System Interlocks **Functional Unit 19.c**				
Power Range Neutron Flux, P-8	≤49.0% RTP	49.0% RTP	≤51.1% RTP	≤49.6% RTP
Functional Unit 19.d				
Power Range Neutron Flux, P-10	≥10.0% RTP	10.0% RTP	≥7.9% RTP	≥9.4% RTP
Functional Unit 19.e				
Turbine Impulse Chamber Pressure, P-13	≤10.0% RTP Turbine Impulse Pressure Equivalent	10.0% RTP Turbine Impulse Pressure Equivalent	≤12.1% RTP Turbine Impulse Pressure Equivalent	≤10.6% RTP Turbine Impulse Pressure Equivalent

Notes:

(a) Minimum Measured Flow is 97,847 gpm/loop

Table 6.7-2 Summary of the Technical Specification ESFAS Setpoint Changes				
	Trip Setpoint		**Allowable Value**	
	Current Value	**SGR/Uprating Value**	**Current Value**	**SGR/Uprating Value**
Containment Pressure – High 1, Safety Injection (and other functions) **Functional Unit 1.c**				
High Setpoint	≤3.0 psig	3.0 psig	≤3.6 psig	≤3.3 psig
Pressurizer Pressure – Low, Safety Injection (and other functions) **Functional Unit 1.d**				
Low Setpoint	≥1850 psig	1850 psig	≥1836 psig	≥1846 psig
Steam Line Pressure – Low, Safety Injection (and other functions) **Functional Unit 1.e**				
Low Setpoint	≥601 psig*	601 psig*	≥578.3 psig*	≥594.5 psig*
Containment Pressure – High 3, Containment Spray **Functional Unit 2.c**				
High Setpoint	≤10.0 psig	10.0 psig	≤11.0 psig	≤10.3 psig
Containment Pressure – High 2, Main Steam Line Isolation **Functional Unit 4.c**				
High Setpoint	≤3.0 psig	3.0 psig	≤3.6 psig	≤3.3 psig
Negative Steam Line Pressure Rate – High, Main Steam Line Isolation **Functional Unit 4.e**				
High Setpoint	≤100 psi	100 psi	≤122.8 psi	≤106.5 psi
Steam Generator Water Level – High-High (P-14), Turbine Trip & Feedwater Isolation **Functional Unit 5.b**				
High-High Setpoint	≤82.4% of span	78.0% of span	≤84.2% of span	≤78.5% of span

Table 6.7-2 (Cont.) Summary of the Technical Specification ESFAS Setpoint Changes				
	Trip Setpoint		**Allowable Value**	
	Current Value	**SGR/Uprating Value**	**Current Value**	**SGR/Uprating Value**
Steam Generator Water Level – Low-Low, Auxiliary Feedwater Functional Unit 6.c				
Low-Low Setpoint	≥38.5% of span	25.0% of span	≥36.5% of span	≥24.5% of span
Steam Line Differential Pressure – High, Auxiliary Feedwater Functional Unit 6.g				
High Setpoint	≤100 psi	100 psi	≤127.4 psi	≤106.5 psi
Safety Injection Switchover to Containment Sump Functional Unit 7.b				
Low Setpoint	≥23.4%	23.4%	≥20.4%	≥22.9%
Loss-of-Offsite Power – 6.9 kV Emergency Bus Undervoltage - Primary Functional Unit 9.a				
Low Setpoint	≥4830 volts with a ≤1.0 sec. time delay	4830 volts with a ≤1.0 sec. time delay	≥4692 volts with a ≤1.5 sec. time delay	≥4800 volts with a ≤1.12 sec. time delay
Loss-of-Offsite Power – 6.9kV Emergency Bus Undervoltage - Secondary Functional Unit 9.b				
Low Setpoint	≥6420 volts with a ≤16 sec. time delay (with Safety Injection)	6420 volts with a ≤13 sec. time delay (with Safety Injection)	≥6392 volts with a ≤18 sec. time delay (with Safety Injection)	≥6392 volts with a ≤14.3 sec. time delay (with Safety Injection)
Low Setpoint	≥6420 volts with a ≤54.0 sec. time delay (without Safety Injection)	6420 volts with a ≤54.0 time delay (without Safety Injection)	≥6392 volts with a ≤60.0 sec. time delay (without Safety Injection)	≥6392 volts with a ≤54.0 sec. time delay (without Safety Injection)

Table 6.7-2 (Cont.) Summary of the Technical Specification ESFAS Setpoint Changes				
	Trip Setpoint		**Allowable Value**	
	Current Value	**SGR/Uprating Value**	**Current Value**	**SGR/Uprating Value**
ESFAS Interlocks **Functional Unit 10.a**				
Pressurizer Pressure, P-11	≥2000 psig	2000 psig	≥1986 psig	≥1996 psig
Functional Unit 10.a				
Pressurizer Pressure, **Not** P-11	≤2000 psig	2000 psig	≤2014 psig	≤2004 psig
Functional Unit 10.b				
Low-Low T_{avg} P-12	≥553°F	553°F	≥549.3°F	≥552.5°F

Notes:

* Time constants utilized in the lead-lag controller for steam line pressure-low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds.

6.8 Anticipated Transient Without Scram

6.8.1 Introduction

The final Anticipated Transients Without a Reactor Scram (ATWS) rule (10CFR50.62) requires the incorporation of a diverse (from the reactor trip system) actuation of the auxiliary feedwater system and turbine trip for Westinghouse designed plants. The installation of the NRC approved ATWS Mitigating System Actuation Circuitry (AMSAC) satisfies this final ATWS rule. The basis for this rule and the AMSAC design are supported by Westinghouse generic analyses documented in NS-TMA-2182 (Reference 1). These analyses were performed based on guidelines published in NUREG-0460 (1978) (Reference 2).

Reference 1 also references WCAP-8330 (Reference 3) and subsequent related documents, which formed the initial Westinghouse submittal to the NRC for ATWS, and which were based on the guidelines set forth in WASH-1270 (Reference 4). HNP is currently licensed on the basis of these calculations and analyses. For SGR/Uprating conditions, the current licensing basis calculations and analyses will be evaluated for continued applicability.

Reference 1 describes the methods used in the analysis and provides reference analyses for two-loop, three-loop, and four-loop plant designs with several steam generator models (including the Model D series) available in plants at that time. The reference analysis results for the four-loop plant design are more limiting than those for the three-loop plant design and demonstrated that the Westinghouse plant designs would satisfy the proposed criteria in NUREG-0460.

The failure of the reactor scram is presumed to be a common mode failure of the control rods to insert into the core. The assumption of this common mode failure is beyond the requirement to address a single failure in the typical FSAR Chapter 15 transient analyses. In addition, the methodology of Reference 1 uses control grade equipment to mitigate consequences of the event and uses nominal system performance characteristics in the evaluation of the event.

6.8.2 Description of Analyses and Evaluations

Reference ATWS Analysis Evaluation

An analysis was performed to evaluate the effect of the HNP Model Δ75 replacement steam generators and power uprate on the reference analysis documented in support of Reference 1. The analysis revised the reference three-loop LOFTRAN model (Reference 5) to include Model Δ75 steam generators and evaluated the Loss of Normal Feedwater and the Loss of Load ATWS events. These are the two most limiting RCS overpressure transients documented in Reference 1. In this analysis, a detailed NOTRUMP computer code (Reference 6) was used to model the Δ75 replacement steam generator responses to transient conditions. The method utilized an iterative analysis technique between the detailed SG model and the LOFTRAN system transient calculations as applied in the analysis supporting Reference 1.

AMSAC Setpoint Evaluation

Additional evaluations were performed to determine the effects of the SGR/Uprating and the adequacy of the AMSAC setpoints. The evaluation for SGR/Uprating effects considered the higher power level, initial primary system average temperature, and RCS flow characteristics. The AMSAC setpoints considered an actuation setpoint at 20 percent of the steam generator narrow range span (5 percent of span below the steam generator low-low level reactor trip setpoint) and a timing delay of 25 seconds. In addition, an evaluation for the arming point (C-20 permissive) was performed.

For these evaluations, the LOFTRAN code was used to bound the SG characteristics that are critical to this event. This model is used for licensing basis calculations for the Loss of Normal Feedwater, Feedline Break, and Steamline Break Mass/Energy release events where uncovering of the steam generator tube bundle is expected to occur. The LOFTRAN model was tuned to provide conservative results in comparison to the Model Δ75 SG characteristics seen in the reference three-loop ATWS transient described previously (Reference ATWS Analysis Evaluation) and was then applied in this evaluation.

Arming Point for AMSAC

The arming point of the AMSAC system was originally recommended to be 40 percent of the nominal power level of the plant. The original basis for this setpoint was to establish a power level below which bulk RCS coolant boiling would not occur for the first 10 minutes of the ATWS transient. A set of sensitivities for the initial power level of the plant were performed based the SGR/Uprating conditions to determine the initial power level that would support this bulk RCS coolant boiling criterion.

6.8.3 Acceptance Criteria

ATWS Rule: The final ATWS rule (10CFR50.62) requires the incorporation of a diverse (from the reactor trip system) actuation of the auxiliary feedwater system and turbine trip for Westinghouse designed plants. The installation of the NRC approved AMSAC design satisfies this final ATWS rule. The bases for this rule and the AMSAC design are supported by Westinghouse analyses documented in Reference 1. The peak RCS pressures reached in these HNP analyses shall be similar to, or less than, the peak pressures reached by the four-loop model peak pressures from Reference 1. The four-loop peak pressures are the most limiting RCS pressures reached and formed the basis for the final ATWS rule.

AMSAC Setpoint Evaluation: Peak pressures reached in the RCS shall remain below the ASME Service Level C Stress limit (3200 psig) proposed in NUREG-0460.

Arming point for AMSAC: The original basis for this setpoint was to establish a power level below which bulk RCS coolant boiling would not occur for the first 10 minutes of the ATWS transient.

6.8.4 Results and Conclusions

Reference ATWS Analysis Evaluation

The transient results demonstrated that the Model Δ75 steam generator three-loop model was less limiting than both the Model 51 steam generator three-loop and four-loop models used in the analysis supporting Reference 1. Therefore, the basis for the final ATWS rule is unaffected by implementation of the Model Δ75 steam generators. This analysis for the Model Δ75 replacement steam generators at the uprated NSSS power level of 2912.4 MWt bounds operation of the plant with the Model Δ75 replacement steam generators at the current power level of 2787.4 MWt.

AMSAC Setpoint Evaluation

The analysis for the SGR/Uprating conditions and the AMSAC specific setpoints demonstrated that the peak pressures reached in the RCS would remain below the ASME Service Level C Stress limit (3200 psig) proposed in NUREG-0460. This analysis for the Model Δ75 replacement steam generators at the uprated NSSS power level of 2912.4 MWt bounds operation of the plant with the Model Δ75 replacement steam generators at the current power level of 2787.4 MWt.

Arming Point for AMSAC

It was determined that an initial power level of 36.5 percent of the uprated power level (2912.4 MWt) would not result in bulk RCS coolant boiling within 10 minutes of the initiation of the transient and thereby satisfies the original criterion established for the AMSAC arming point. The 36.5 percent of 2912.4 MWt is based upon a set of sensitivity studies to determine the power level at which the original criterion would be satisfied. An arming point value less than or equal to 36.5 percent of 2912.4 MWt would also satisfy the criterion. The peak pressures reached during these transients were significantly below the full-power analysis cases. This recommended arming point for the AMSAC compensates for the revised conditions of the plant.

6.8.5 References

1. Westinghouse Letter NS-TMA-2182, "Anticipated Transients Without Scram for Westinghouse Plants," December 1979.

2. NRC Staff Report NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," April 1978.

3. WCAP-8330, "Westinghouse Anticipated Transient Without Trip Analysis," August 1974.

4. NRC Report WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water Cooled Power Reactors," September 1973.

5. Westinghouse WCAP-7907-P-A (Proprietary), "LOFTRAN Code Description," WCAP-7907-A (Nonproprietary), April 1984.

6. Westinghouse WCAP-10079-P-A (Proprietary), "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

7.0 NUCLEAR FUEL

7.1 Core Thermal-Hydraulic Design

This analysis is outside the Westinghouse scope of supply on the SGR/Uprating program. This information will be provided by CP&L or other vendors.

7.2 Fuel Core Design

This analysis is outside the Westinghouse scope of supply on the SGR/Uprating program. This information will be provided by CP&L or other vendors.

7.3 Fuel Core Design and Performance

This analysis is outside the Westinghouse scope of supply on the SGR/Uprating program. This information will be provided by CP&L or other vendors.

7.4 Heat Generation Rates

7.4.1 Introduction and Background

Gamma ray heat generation rates in the lower core plate were specifically determined at the SGR/Uprating conditions; heat generation rates for other reactor internals components were obtained through a scaling process. The heat generation rate values were supplied as input for use in the reactor internals structural evaluations described in Section 5.2.

The presence of heat generated in the reactor internals components, along with a distribution of fluid temperatures, results in thermal gradients within and between components. These temperature gradients result in thermal stresses and thermal growth that must be accounted for in the design and analysis of the various reactor internals components. The primary design considerations are (1) to ensure that thermal growth is consistent with the functional requirements of components and (2) to ensure that the applicable ASME Code requirements are satisfied. In order to satisfy these requirements, the reactor internals components must be analyzed with respect to fatigue and maximum allowable stress considerations.

The reactor internals components that are subjected (either directly or indirectly) to significant heat generation effects are the upper and lower core plates, the lower core support, the core baffle plates, the former plates, the core barrel, the neutron pad, the baffle-former bolts, and the barrel-former bolts. Note, however, that the upper core plate, the lower core support, and the neutron pad experience little, if any, temperature rise over the surrounding reactor coolant due to relatively low heat generation rates (generally less than 50 Btu/hr-lbm).

WCAP-9620, Rev. 1 (Reference 1) provides heat generation rates for Westinghouse plants, including HNP, that were intended to be applicable for the life of the plants. However, the introduction of low leakage loading patterns has indicated that plant-specific analyses may be required for the lower core plate to assess the increased power in the radial center region of the core. Conversely, it should be noted that the power in the (radial) core periphery is reduced for low leakage loading patterns which provides margin to the WCAP-9620, Rev. 1 heating rates for components that are radially outward from the center of the core.

This section describes how the heat generation rates were determined for the HNP lower core plate at the SGR/Uprating conditions and how the component average heating rate values were determined for the remaining reactor internals components.

7.4.2 Reactor Internals Heat Generation Rates - Lower Core Plate

7.4.2.1 Description of the Analysis

Power distribution and Siemens Power Corporation (SPC) fuel geometry data has been acquired for the equilibrium reload fuel cycle and evaluated relative to data Westinghouse has previously incorporated into the heat generation rate methodology.

Conservative axial and radial power distributions were assumed in the long-term heat generation rate calculation. For the short-term bottom-peaked calculation, the axial power distribution was taken from the Core Radiation Source Data (CRSD) document as described in Reference 1. A conservative radial power distribution was used for the short-term heat generation calculation. HNP specific radial (assembly-by-assembly) loading patterns and short-term and long-term axial power distributions were also analyzed.

The analysis was performed through the use of the DORT (Reference 2, Version 3.1) discrete ordinates code, which has been used by Westinghouse to calculate heating rates for other projects that have been approved by the NRC. The lower core plate was analyzed in an R-Z (cylindrical) geometry calculation based on the equivalent volume cylindrical core concept. The varying amounts of structure located axially below the core were approximated as a number of regions each with the appropriate amount of stainless steel, water, and other materials uniformly homogenized throughout the region. The R-Z geometry included the lower three feet of the core and extended axially to one foot below the lower core plate and radially out to the inner radius of the reactor vessel.

Varied cases were used to determine the HNP lower core plate heat generation rates at SGR/Uprating conditions. Westinghouse Vantage 5H (V5H) fuel was used as the base case, but selected Siemens data for the fuel bottom nozzle and fuel rod end plug were evaluated in determining which calculation cases should be selected for the final (limiting) heat generation rates. These limiting cases were then used in the reactor vessel internals analysis.

7.4.2.2 Acceptance Criteria for Analyses

There are no acceptance criteria for this analysis, since the applicable design considerations for the reactor internals components are evaluated in subsequent calculations that use these heat generation rate results as inputs.

7.4.2.3 Results

The heat generation rates for the lower core plate were provided as input for the reactor vessel internals analysis described in Section 5.2.

Long-Term Case

The average long-term heat generation rate in the central portion of the core plate (outer radius - 59.8 inches) is 528.1 Btu/hr-lbm. In the outer annular portion (outer radius - 66.9 inches), the average is 62.7 Btu/hr-lbm.

Short-Term Case

With respect to the short-term axial power distributions, the WCAP-9620, Rev. 1 values are more conservative than the corresponding HNP data and will result in higher heat generation rates in the lower core plate.

With respect to the short-term radial distribution, a conservative radial power distribution was used for the short-term heat generation calculation.

The average short-term heat generation rate is 1403.3 Btu/hr-lbm in the inner portion of the core plate and 181.4 Btu/hr-lbm in the outer portion.

7.4.3 Radial Internals - Core Barrel, Baffle Plates, Neutron Pad

7.4.3.1 Description of the Analyses

Design basis heat generation rates that are applicable to the HNP radial internals are contained in Appendices H and I of Reference 1. The core power distributions upon which these calculations were based were derived from 25 independent fuel cycles in 11 three-loop reactors and represented an upper tolerance limit of beginning-of-cycle (BOC) and end-of-cycle (EOC) power in the peripheral fuel assemblies, based on a 95 percent probability with 95 percent confidence. The peripheral fuel assemblies are defined as those with one or two faces or one corner adjacent to the core baffle. Most of the 25 fuel cycles were out-in loading patterns that, when combined with the statistical processing selected, resulted in a core power distribution that was biased high on the core periphery. This high bias was desired by the reactor internals analysts to ensure conservative, but not unrealistic, results in the critical baffle-barrel region of the reactor internals.

Utility interest in reducing the rate of PWR reactor vessel embrittlement by reducing the incident fast neutron flux to the reactor vessel through fuel management and core periphery modifications has grown in recent years. In addition, the fuel cycle cost advantages of reduced core neutron leakage coupled with higher permissible core power peaking limits have resulted in fuel management strategies with significantly lower power levels in the peripheral fuel assemblies than was the case with traditional out-in fuel management.

The evaluation of heat generation rates for the radial internals used HNP-specific assembly-wise core power distributions for the power uprate and CRSD power distributions from Reference 1 (Figure A-3).

An assessment was made of the effect of the core power distributions on the heat generation rates in the core baffle plates and core barrel. The approach taken was to use scaling factors that account for the facts that (1) heat generation rates in the radial internals regions are the result of radiation leakage from the periphery of the core, and (2) to a close approximation, the heat generation rate in a given region is proportional to the power produced in adjacent fuel regions. These ratio expressions were determined based on discrete ordinates transport theory calculations using various core power distributions.

For the long-term axial power distribution, inputs from HNP were evaluated versus the Reference 1 long-term axial power distribution; the HNP data for long-term axial power distributions is less limiting.

For the short-term axial power distributions, comparisons show that the HNP data is less limiting than the Reference 1 axial power distributions in the regions of the core where power is being maximized for analysis.

7.4.3.2 Acceptance Criteria for Analyses

There are no acceptance criteria for the heat generation rate analysis, since the applicable design considerations for the reactor internals components are evaluated in subsequent calculations that use these heat generation rate results as inputs.

7.4.3.3 Results

For consideration of baffle plate heating, core barrel heating, and neutron pad heating, the radial power and axial power distributions were evaluated, and it was concluded that the Reference 1 heat generation rates apply (i.e., are more limiting).

Changing core power distributions affects reactor internals heat generation rates. The effect of implementing low leakage loading patterns has been examined relative to the reactor internals design basis, which assumed traditional out-in fuel management. In general, the change reduces the baffle-barrel heat generation rates by one-third to one-half. The Reference 1 distributions through the thickness of the baffle plates and core barrel are not expected to be significantly affected by the change from out-in to low leakage loading patterns.

The additional values are designed to augment the values reported in Reference 1, providing additional information on the heating rate effects of current, realistic power distributions. However, in order to retain the conservatism associated with the work performed and documented in Reference 1, it is necessary to continue to use the absolute magnitudes described in that document.

The heat generation rates for the radial components were provided as input for the reactor vessel internals analysis described in Section 5.2.

7.4.4 References

1. "Reactor Internals Heat Generation Rates and Neutron Fluences," WCAP-9620, Revision 1, A. H. Fero, December 1983.

2. RSICC Computer Code Collection CCC-650, "DOORS3.1 One-, Two-, and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," August 1996.

7.5 Neutron Fluence

This analysis is outside the Westinghouse scope of supply on the SGR/Uprating program. This information will be provided by CP&L or other vendors.

7.6 Source Terms

7.6.1 Introduction

7.6.1.1 Source Term Calculation

Source terms for accident and normal operating conditions were determined for the SGR/Uprating. The results are provided primarily for input to dose analyses and shielding evaluations. The reanalysis for the SGR/Uprating provides radiation sources for the following areas:

- Maximum Credible Accident
- Fuel Handling Accident
- Design (Reactor Coolant System [RCS]) Radiation
- Design Basis Secondary Normal Plant Operation
- N-16 Activity
- Pressurizer
- Solid Radwaste and Evaporator Concentrates
- Gas Decay Tank and Volume Control Tank
- Tritium
- Environmental Qualification
- Decay Heat

In general the power level used for calculations that are accident or "design" conditions is 2958 MWt, which is the SGR/Uprating core power plus two percent for uncertainty in determining power level. However, there are instances where the nominal 2900 MWt power level is used; this is for "normal" source calculations where the basis is ANS/ANSI-18.1 and includes cases for N-16 and tritium calculations where the two percent is small compared to other variables in the analysis.

The effect of higher power is normally to increase calculated source terms. However, enrichment and cycle length are also factors, which, if changed from previous analyses, can also affect source terms on a nuclide-by-nuclide basis to either increase or decrease the individual nuclide activity.

In general, there is a direct proportionality between power level and fission product activity, which can be accounted for by scaling. Short-lived activity is proportional to power, whereas long-lived activity is proportional to power multiplied by time (burnup).

Higher enrichment results in more U-235 fission and lower thermal flux. Conversely, thermal flux is higher for lower initial enrichments at the same burnup, and thermal flux generally increases over the length of a cycle. Initial enrichment is a particularly significant parameter in explaining the thermal flux-related removal term for Xe-135. A higher thermal flux results in a higher removal rate for Xe-135, so an increasing thermal flux has the effect of decreasing the relative quantity of that nuclide.

Higher burnup results in more Pu-239 fission, higher activities of long-lived fission products and higher thermal flux. Burnup-dependent behavior includes decreasing fission yield for kryptons and bromines, attributable to plutonium build-in. In general, the effects of increasing burnup can be estimated by direct scaling of long-half-lived nuclides.

The above effects of power, burnup, and enrichment result in a complicated dependence of each nuclide on changes in these parameters; however, all are included in the computer analysis.

Changes to fuel mass affect results by means of a direct relationship to fission yields and thermal flux considerations. The effects are similar to those discussed for initial enrichment. That is, on the first order, an increase in mass is equivalent to an increase in enrichment.

The most significant "coolant activation product" during operation is N-16 (Reference 1); the activity for this nuclide will be directly proportional to the increase in power levels. This activity also varies with the time that reactor coolant takes to pass through the core and the transit time around the reactor coolant loop. The situation for activated corrosion products is similar. In this case, however, there is some degree of depletion or target burnout that may ameliorate this increase.

7.6.1.2 Computer Codes Used

Fission product inventories and decay heat were modeled with ORIGEN2, Version 2.1 (Reference 2). ORIGEN2 is a versatile point-depletion and radioactive-decay computer code for use in simulating nuclear fuel cycles and calculating the nuclide compositions and characteristics of materials contained therein. The ORIGEN2 code is an industry standard code based on the latest industry experimental data. In general the data are up-to-date, well documented, and accepted by the industry, and therefore, they are appropriate for the SGR/Uprating analyses.

7.6.2 Maximum Credible Accident Sources

7.6.2.1 Introduction

The HNP SGR/Uprating will provide a higher power rating for the plant and include, concurrently, the installation of new steam generators. The purpose of this report section is to present the maximum credible accident radiation sources for the uprated power level.

Calculations were performed to include variations in the power level and fuel management parameters. The selection of source terms from multiple cases is made to provide a bounding set of isotopics for use in dose calculations based on the assumptions of the Technical Information Document, TID-14844 (Reference 3).

7.6.2.2 Description of Analyses and Evaluations

Fuel management for the equilibrium cycle, a long equilibrium cycle, and a short equilibrium cycle was considered. The intent of using varied fuel management schemes for the analysis is to encompass variations that may occur in plant operation.

Calculations were made to determine the maximum credible accident radiation sources for the uprated power level. Releases are based on the release fractions described in TID-14844, 50 percent of halogens, 100 percent of noble gases, and one percent of remaining fission products, gap release fractions of Regulatory Guide 1.25 (Reference 4), release fractions of Regulatory Guide 1.4 (Reference 5) and NUREG/CR-5009 (Reference 6).

In addition, the TID-14844 released gamma energy is integrated over one year and combined with the containment free volume to provide a dose curve for confirming the Equipment Qualification (EQ) limit.

ORIGEN is a computer code system for calculating the buildup, decay, and processing of radioactive materials. ORIGEN2.1 (Reference 2) is used to calculate the fission product inventory with input describing the fuel enrichment, fuel burnup, power level, and fuel masses comprising the core. Core activities for accident source terms are taken directly from ORIGEN2.1.

Variations modeled provide a range of nuclide activities possible including an equilibrium cycle, a long equilibrium cycle, a short equilibrium cycle, axial blanket implementation, and achieving burnup at reduced power. From these calculation cases, maximum nuclide activities were selected to provide a bounding source.

7.6.2.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the Balance of Plant [BOP] Licensing Report).

7.6.2.4 Results

The results of the nuclide release calculations are used as input to the dose rate calculations. The integration of the TID-14844 release may be used to evaluate the gamma EQ radiation environment.

7.6.3 Fuel Handling Accident Sources

7.6.3.1 Introduction

This section presents the Fuel Handling Accident radiation sources for the SGR/Uprating program.

Fuel inventories are taken from equilibrium cycles with nominal, long, and short cycle lifetimes. Factors applied to inventories are based on Regulatory Guide 1.25 (Reference 4), the appropriate peaking factor for HNP, and applicable NUREG (Reference 6) data.

Using the calculated results for fission product inventory, the factors applied provide releases at 100 hours after shutdown.

7.6.3.2 Description of Analyses and Evaluations

Fuel management for the equilibrium cycle, a long equilibrium cycle, and a short equilibrium cycle was considered. The intent of using varied fuel management schemes for the analysis is to encompass variations that may occur in plant operation.

A two-fold approach was used to determine the inventory to be released in the fuel handling accident. First, a single assembly with the maximum inventory at shutdown was found for the fuel management schemes. Core activities for accident source terms were taken directly from ORIGEN2.1, which was used to calculate the fission product inventory with input describing the fuel enrichment, power level, cycle times, and fuel masses comprising the core.

In the second approach, the average assembly isotopic inventory was determined for the nominal equilibrium cycle at shutdown by dividing the whole core isotopic inventory by the number of assemblies in the core. Multiplying the average inventory by the peaking factor, it was shown that this inventory is greater than that of the maximum-inventory single assembly from the fuel management schemes and, therefore, provides a conservative basis for release calculations.

The Regulatory Guide 1.25 gap release factors were applied to the inventory of the average assembly at the end of the nominal equilibrium cycle. The gap inventory was then decayed for 100 hours. Finally, the peaking factor was applied to the inventory.

Nuclides that were not included in the inventory include those with low activity (I-129) and those with short half-lives (I-134, Kr-85m, Kr-87, Kr-88, Kr-89, Xe-137, and Xe-138) that have decayed to negligible levels at 100 hours after shutdown.

All ORIGEN2.1 runs used for inventories were made at 102 percent of 2900 MWt, which is consistent with thermal power uncertainties identified for HNP.

For the Fuel Handling Building, the inventory was increased to reflect 314 rods damaged in the accident. Finally, activities that include release fractions and peaking factor application for 52 BWR assemblies in the spent fuel pit that might be affected in a fuel handling accident were added to the activities calculated for the PWR fuel.

7.6.3.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.3.4 Results

The results of the nuclide release calculations are input to dose rate calculations.

7.6.4 Reactor Coolant System Radiation Sources

7.6.4.1 Introduction

This report section provides RCS radiation sources for the HNP SGR/Uprating.

For the reactor coolant system, maximum coolant activities obtained during a cycle of operation are calculated. In the calculation of maximum coolant activities obtained during a cycle of operation, small cladding defects in fuel that generate one percent of the core power are assumed to be present in each core loading and uniformly distributed throughout the core.

7.6.4.2 Description of Analyses and Evaluations

Fuel management for the equilibrium cycle, a long equilibrium cycle, and a short equilibrium cycle was considered. Also considered were fuel designs that reflect the highest enrichment, as was the use of axial blankets. The intent of using varied fuel management schemes for the analysis encompasses variations that may occur in plant operation.

Parameters in the calculation of the reactor coolant fission product activities include the pertinent information concerning the expected coolant cleanup flow rate and demineralizer effectiveness. The minimum RCS was used and is conservative for the calculations that provide activities in microcuries per gram (μCi/g) of coolant.

It was assumed that power produced by one percent of the fuel comes from fuel with fuel defects; this is the standard basis for design sources. Also, there is no purging of the Volume Control Tank (VCT) during operation, which conservatively increases RCS activities and is consistent with Radiation Analysis Manual calculations.

The applicability of the calculation extends to 2958 MWt, which is a 2900 MWt nominal core power increased by two percent to allow for uncertainty in determining power level.

For the reactor coolant system, maximum coolant activities obtained during a cycle of operation were calculated. The ORIGEN2.1 code (Reference 2) is used to irradiate fuel through burnups attained by each fuel region in the cycle in order to determine activity inventories in the core. Inventory at intervals from zero burnup through the discharge burnup are used to determine the maximum activities that occur in the cycle.

In these calculations, small cladding defects in fuel that generates one percent of the core power are assumed to be present at initial core loading and uniformly distributed throughout the core. Similar defects are assumed to be present in all reload regions.

7.6.4.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.4.4 Results

The results of the nuclide release calculations are input to dose rate calculations or are used as input to shielding calculations.

7.6.5 Design Basis Secondary Sources

7.6.5.1 Introduction

This report section provides the design bases steam generator secondary side radiation sources for the HNP SGR/Uprating and revised RCS volume.

7.6.5.2 Description of Analyses and Evaluations

The steam generator blowdown processing system maintains the water effluent from the steam generators at a chemical and radiological specification suitable for its recycle into the main condenser or for its discharge. During normal operation, fluid from each steam generator enters under pressure into a blowdown heat exchanger, where the temperature is reduced and the blowdown is directed through the prefilter and mixed-bed demineralizers in series. After passing through outlet filters, the fluid flows through a radiation monitor and would normally be recycled to the feedwater flow, returning to the steam generator, but may be discharged to the main condenser. If the main condenser inventory reaches a high level it can be pumped to either the condensate storage tank or discharged to the environment.

The RCS radiation sources discussed in Section 7.6.4 were used as input to the calculation of steam generator secondary side activities. A value of one (a high value) is assumed for primary to secondary reactor coolant leakage. This assumption gives conservatively high values of input to the secondary side fluid.

Calculations were performed to determine the following:

- Radionuclide concentrations in the secondary side water and steam in a PWR steam generator given a primary-to-secondary leak and the operation of a steam generator blowdown processing system for cleanup
- The buildup of activity in the blowdown processing system demineralizer and filter
- The gamma ray sources for the blowdown water (secondary side water), resin and filter activities

The applicability of the calculation extends to 2958 MWt, which is a 2900 MWt nominal core power increased by two percent to allow an uncertainty in determining power level. The selection of source terms from multiple cases is made to provide a bounding set of isotopics for the secondary side given the plant and system operation parameters used.

7.6.5.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.5.4 Results

The results of the nuclide release calculations are used as input to dose rate calculations or are used as input to shielding calculations.

7.6.6 Normal Plant Operation Source Terms

7.6.6.1 Introduction

The normal plant operational source terms establish the long-term concentrations of principal radionuclides in the fluid streams of the plant for subsequent use in predicting the expected release of radioactive materials from various effluent streams. The fluid streams of the plant are the reactor primary coolant and the steam generator water and steam.

Normal plant operation source terms are based on the American National Standard (ANS) Source Specification, ANSI/ANS-18.1-1984, "Radioactive Source Term for Normal Operation of Light

Water Reactors" (Reference 7). The purpose of the standard, ANSI/ANS-18.1-1984, is to provide for a uniform approach, applicable to light-water-cooled nuclear power plants, for the determination of expected concentrations in fluid streams. Through application of this standard, a common basis for the determination of radioactive source terms is established with the goal of providing a consistent approach for those involved in the design, licensing, and operation of nuclear power plants.

The numerical values given in the ANSI/ANS-18.1-1984 standard are based on available data from operating plants that use Zircaloy-clad, uranium-dioxide fuel. Normal plant operational sources for HNP are established by appropriate scaling of standard values to define source term values specific to the normal plant operating parameters. The scope of the calculated values for normal plant operation sources are the values and algorithms included in the ANSI/ANS-18.1-1984.

7.6.6.2 Description of Analyses and Evaluations

The normal plant operation source term analysis uses the ANSI/ANS-18.1-1984 specifications and formulations for calculating the radionuclide activity in the fluid streams of a Light Water Reactor (LWR) nuclear plant. The use of this standard is the accepted industry methodology for performing these calculations. The ANSI/ANS-18.1-1984 data is scaled to the proper thermal power level and specific inputs related to HNP SGR/Uprating conditions.

If the plant-specific parameters such as thermal power level, fluid system flow rates, and system fluid quantities are the same as the ANS/ANSI-18.1-1984 standard values, the source-term values of the standard are used without modification. In the case where any of the parameters differ from the values used in the standard, the source term values must be modified to account for these differences by using adjustment factors specified in ANSI/ANS-18.1-1984.

The ANSI/ANS-18.1-1984 standard is an update of the American National Standard ANSI N237 (Reference 8) based on a compilation of available operating plant data concerning primary coolant concentrations, steam generator tube leakage, and secondary side radionuclide behavior. The adjustment factors and procedures for effecting adjustments in the calculations are based on methods in ANSI N237 and retained in ANSI/ANS-18.1-1984. NUREG-0017, Revision 1 (Reference 9), uses the data values, adjustment factors, and methods in ANSI/ANS-18.1-1984. Therefore, the use of ANSI/ANS-18.1-1984 is appropriate for the update of HNP normal plant operation sources.

The calculations performed predict concentrations of the noble gases, halogens, rubidium, cesium, N-16, tritium, and other radioisotopes in the various fluid streams of HNP for normal plant operation including anticipated operational occurrences. The list of radionuclides and the concentration values predicted in this analysis are based on the current standard, ANS/ANSI-18.1-1984 (Reference 7) and are consistent with NUREG-0017, Revision 1 (Reference 9).

7.6.6.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.6.4 Results

The results of the nuclide release calculations are used as input to dose rate calculations or are used as input to shielding calculations.

7.6.7 Reactor Coolant N-16 Activity

7.6.7.1 Introduction

For the SGR/Uprating program, the RCS volumes and fluid system component coolant masses for the HNP have been updated to reflect the installation of the Model Δ75 replacement steam generators. The N-16 specific activities and loop transit times in the RCS are updated to reflect the SGR changes, as well as the uprating in power level. The predicted N-16 specific activity in the RCS is based on predicted O-16 reaction rates, primary system component volumes, and RCS temperature and fluid flow conditions for HNP.

7.6.7.2 Description of Analyses and Evaluations

The N-16 neutron activation rates or O-16 neutron reaction rates, i.e. the product of neutron flux and reaction cross-section for the $^{16}O(n,p)^{16}N$ neutron reaction, employed in the analysis are those developed for a three-loop plant with a 1.125-inch thick baffle neutron pad at a reactor thermal power of 2652 MWt. The HNP reactor internals geometry is the same as the internals geometry used in the analysis, and reaction rates are directly applicable to HNP with the only correction a direct scaling of the input data to the HNP uprated core power level of 2900 MWt.

Primary loop component volumes are used to predict primary loop transit times for the calculation of the N-16 specific activity as a function of location in the RCS loop. Volumes of the various components in the primary system are scaled to hot operating conditions for the N-16 analysis.

Calculations were performed to determine the radionuclide source terms from the buildup and decay of radioactive materials in the fluid systems. The N-16 specific activity calculation uses the primary system volumes, flow rates, flow fractions, and coolant densities at operating conditions to predict the reactor coolant N-16 specific activity at specified locations in the primary loop. Calculations of the N-16 specific activity in the pressurizer liquid and vapor volumes are also predicted in the analysis.

7.6.7.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.7.4 Results

The results of the nuclide release calculations are input to dose rate calculations or are used as input to shielding calculations.

7.6.8 Pressurizer Sources

7.6.8.1 Introduction

This report section provides the design basis pressurizer radiation sources for the SGR/Uprating and revised RCS volume.

7.6.8.2 Description of Analyses and Evaluations

The RCS sources discussed in Section 7.6.4 are used as inputs to calculations for the pressurizer sources. The RCS sources are calculated for one percent fuel defects and a core power level of 2958 MWt. The applicability of the calculation extends to 2958 MWt, which is a 2900 MWt nominal core power increased by two percent to allow an uncertainty in determining power level. The selection of input terms is made from multiple cases of RCS activity calculations to provide a bounding set of isotopics for the pressurizer activity calculation.

The pressurizer liquid specific activity is assumed to be the same as that of the reactor coolant. Pressurizer steam phase radiogas concentrations are based on the stripping of radiogases from the continuous 2-gpm pressurizer spray and the subsequent buildup of these radiogases in the steam space. The buildup time is assumed to be one year. The radiogases are assumed to be completely stripped from the spray, except for Kr-85, where a stripping fraction of 0.9 is used.

Pressurizer steam phase iodine concentrations are obtained from the liquid phase nuclide activities and measured values of the partition coefficient for I-131. A large partition coefficient was chosen to maximize the activities. It was assumed to apply to all radioiodines.

7.6.8.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.8.4 Results

The results of the nuclide release calculations are used as input to dose rate calculations or are used as input to shielding calculations.

7.6.9 Solid Radwaste And Evaporator Concentrates

7.6.9.1 Introduction

This report section provides the solid waste and evaporator concentrates radiation sources for the SGR/Uprating program. Sources are generated both for normal operation and design basis.

The radiation sources are presented for the boron recycle evaporator concentrates and for demineralizers in the CVCS and boron recycle system. Other waste evaporators and demineralizers were not updated since their function has been replaced by other equipment.

7.6.9.2 Description of Analyses and Evaluations

The boron recycle evaporator bottoms activities were calculated. The activity of the CVCS mixed-bed demineralizer resins for design basis sources was calculated. The other demineralizer resin activities for both normal and design activities were also calculated. Activity on each resin was calculated using system flow rates and decontamination factors to determine the rate of radioactivity deposit on the resin. Since the cation bed is only used intermittently (between 0 and 10 percent of the time), for the mixed bed, the conservative assumption is made that all the long-lived activity is removed on the mixed bed. For the maximum activity on the cation bed, it was assumed that short-lived cations will be equal in activity to the mixed bed, and for the long-lived cations a maximum of 10 percent of the total mixed bed activity will be present.

7.6.9.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.9.4 Results

The results of the nuclide release calculations are used as input to dose rate calculations or are used as input to shielding calculations.

7.6.10 Gas Decay Tank and Volume Control Tank Sources

7.6.10.1 Introduction

This report section provides the Gas Decay Tank (GDT) radiation sources and the Volume Control Tank (VCT) radiation sources at the SGR/Uprating conditions.

GDT radiation sources were determined for the shutdown of the reactor after operation for an equilibrium cycle and for a 40-year inventory of Kr-85. VCT radiation sources were determined for maximum activities of nuclides in the VCT during operation of the equilibrium cycle.

7.6.10.2 Description of Analyses and Evaluations

Gaseous Waste Processing System (GWPS) radiation sources are calculated as GDT inventory after 40 years of operation for both the design basis RCS sources and normal RCS sources. For design basis RCS activities, the RCS activities were developed with and without the operation of the GWPS (purging of the VCT during operation).

Liquid Waste Processing System (LWPS) radiation sources in FSAR Table 15.7.2-2 (and repeated in Table 15.7.2-3) are for non-seismic equipment for consideration of liquid system equipment failure and release of radioactive noble gases.

RCS and VCT activities are used as input to the GDT and VCT calculations. These are taken from an equilibrium cycle, a long equilibrium cycle, a short equilibrium cycle, high enrichment, and axial blanket implementation. There is no purging of the VCT during operation. This conservatively increases the reactor coolant and VCT activities relative to operation with purging during operation.

GDT radiation sources are calculated for shutdown of the reactor after operation for an equilibrium cycle. Maximum RCS and VCT activities for an operating cycle are taken as input.

Since no purging of the VCT has occurred in the cycle there is an inventory of noble gases in the VCT vapor. A calculation is done to simulate the purging of the VCT to the GDT at shutdown, degassing of the RCS with the maximum RCS letdown rate for 3 hours, followed by another purge to the GDT. The cycle of degassing for 3 hours and purging to the GDT is continued for a total of 10 purges. For the iodines, there is assumed a decontamination factor (DF) of 10 and a partition factor of 100 to determine the activity in the VCT vapor.

Maximum GDT (except for Kr-85) activities are selected over the degassing period for the GDT sources. The method takes this approach to maximize the activities for input to accident analysis.

The treatment of Kr-85 is different from the other isotopes. This calculation assumes that the entire RCS Kr-85 inventory produced in the cycle is situated in the GDT.

Noble gases and iodines are the nuclides of interest for the analysis of the VCT rupture. Calculations were done by using maximum RCS and VCT activities and applying factors to account for reductions in concentrations due to resin bed removal or VCT stripping.

For Kr-85, Henry's Law constant is used to determine the Kr-85 in Henry's law equilibrium with the RCS. Some small amount of noble gases is in the VCT liquid. This amount was calculated by using the RCS activity and the stripping fraction for the nuclide.

The majority of the iodines are in the VCT liquid. The specific activities, μCi/g, are calculated by applying a DF of 10 to the RCS values in order to account for removal by the mixed bed demineralizer that is upstream of the VCT.

The GWPS accumulated inventory values based on design RCS activities after one cycle are calculated by multiplying the GDT activity by the volume of the control tank, except for Kr-85. The decay of the nuclides during operation and shutdown from cycle to cycle reduces the GWPS inventory values to very small values so that the last cycle of operation provides the GWPS inventory.

Using the values of inventory, the decay of nuclides is calculated for 30 days and 50 days after shutdown. The calculation of the expected (using normal sources) accumulated radioactivity in the GWPS after forty years of operation is done by the same method, using one-cycle inventory and the 40-year buildup of Kr-85.

7.6.10.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since the radiological consequences/shielding are evaluated in subsequent calculations that use these sources as inputs (see the BOP Licensing Report).

7.6.10.4 Results

The results of the nuclide release calculations are used as input to dose rate calculations or are used as input to shielding calculations.

7.6.11 Tritium Source Calculations

7.6.11.1 Introduction

RCS coolant volumes for HNP have been updated to reflect the SGR/Uprating. Tritium generation in the RCS was updated to reflect the SGR changes, the uprating in power level, and

the projected refueling plan for HNP. Analyses were performed to predict the tritium generation and distribution in the reactor coolant system based on the SGR/Uprating and projected operational plans at the HNP.

7.6.11.2 Description of Analyses and Evaluations

Tritium is produced in Light Water Reactors in several ways. The primary method is a ternary fission product from fission in the fuel rods of the active core.

Tritium from this source, along with tritium produced from boron reactions in burnable poison rods and fuel rods containing boron as a burnable absorber, must diffuse through the fuel or burnable poison material and cladding before release to the RCS. Tritium is also produced directly in the reactor coolant through nuclear reactions involving boron, lithium, and deuterium.

The utilization of burnable poison rods or fuel rods containing boron is not projected for HNP in fuel cycles beyond cycle 7, therefore, analysis of the tritium source from burnable poisons is not included in this analysis.

The analysis of the tritium sources is based on either the thermal power in the active core or the neutron flux levels in the coolant water of the active core.

The tritium source from ternary fission is based on the thermal power level of the reactor and the fission product yield for the tritium isotope. Subsequent release of the tritium to the RCS coolant water is based on either design basis or expected basis release fractions of tritium from Zircaloy-clad fuel rods.

Tritium sources from neutron reactions with elements in the coolant water are based on the projected levels of boron and lithium added to the coolant water during normal operation or the naturally occurring deuterium in the coolant water. The analyses for the three sources of tritium in the coolant water, i.e., coolant soluble boron used for reactor control, coolant soluble lithium used for pH control of the coolant water, and coolant deuterium are based on the same methodology. Neutron flux values in the coolant water regions of the active core are obtained by scaling the core-average flux values by flux ratios of region-wise fluxes from unit cell for a representative fuel pin lattice. The group-wise scaling factors correct the core-average fluxes to coolant water region fluxes. Boron, lithium, and deuterium reaction rates in the coolant water of the active core are predicted by multiplying the neutron fluxes by the reaction cross sections for the tritium producing reactions of boron, lithium, or deuterium isotopes in the coolant water.

7.6.11.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since subsequent calculations are performed for tritium releases that use these sources as inputs (see the BOP Licensing Report).

7.6.11.4 Results

Results of the tritium source analysis are used to evaluate plant tritium generation and release.

7.6.12 Decay Heat Generation

7.6.12.1 Introduction

This report section provides decay heat generation for the SGR/Uprating and revised RCS volume. Decay heat is calculated for shutdown of the reactor after long-term, steady-state operation for an equilibrium cycle at intervals useful in fluid system analysis and plant procedures.

7.6.12.2 Description of Analyses and Evaluations

The anticipated HNP fuel management strategies, which include an equilibrium cycle, a long equilibrium cycle, and a short equilibrium cycle, were considered. The intent of using varied fuel management schemes for the analysis is to encompass variations that may occur during plant operation. The analysis provides a bounding decay heat curve.

ORIGEN2.1 runs for the equilibrium cycle(s) are used to calculate a total core inventory of actinides and fission products (Reference 2). The ORIGEN2.1 calculations were performed at two-percent power above the rated power of 2900 MWt, or 2958 MWt. Decay heat values were taken directly from ORIGEN2.1.

After the reactor is tripped, fissioning of considerable magnitude continues due to delayed neutrons for a brief time, but rapidly diminishes (after about 100 seconds) to insignificant relative to the heat produced by fission product and actinide decay. The period of interest for fluid systems analysis in calculating the Residual Heat Removal (RHR) cooldown transient is from about 4 hours after reactor shutdown to about 50 hours. Therefore, residual heat due to delayed neutron fissioning is not accounted for in this analysis since the first decay step in the ORIGEN2.1 runs documented here is one hour after shutdown.

7.6.12.3 Acceptance Criteria

There are no specific acceptance criteria for these calculations since subsequent calculations use the bounding decay heat curve as inputs (see the BOP Licensing Report).

7.6.12.4 Results

Decay heat was calculated for shutdown of the reactor after long-term, steady-state operation for an equilibrium cycle at intervals useful in fluid system analysis and plant procedures.

7.6.13 Source Term Analysis - Conclusions

Source terms for accident and normal operating conditions were determined for the SGR/Uprating. The results are provided primarily for input to dose analyses and shielding evaluations.

In general, the power level used for calculations that are accident or design conditions is 2958 MWt, which is the SGR/Uprating (core) power plus two percent for uncertainty in determining power level. However, there are instances where the nominal 2900 MWt power level is used; this is for "normal" source calculations where the basis is ANSI/ANS-18.1 and includes cases for N-16 and tritium calculations where the two percent is small compared to other variables in the analysis.

Whether for design or normal radiation source calculations, the effect of higher core power is to increase calculated source terms. Therefore, the results obtained with the Δ75 replacement steam generators at the uprated NSSS power of 2912.4 MWt (core power of 2900 MWt) bound operation with the Δ75 replacement steam generators at the current NSSS power of 2787.4 MWt (core power of 2775 MWt).

Since radiation source terms are provided for input to dose analyses, shielding analyses or evaluations, or for plant use, the users of the source terms must determine, when appropriate, that the resulting analysis or evaluation is consistent with and continue to comply with the current HNP licensing.

7.6.14 References

1. *Nuclear Reactor Engineering*, Sections 6.189, 7.36 and 7.37, Samuel Glasstone and Alexander Sesonske, 1963.

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3. Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," J. J. DiNunno, F. D. Anderson, R. E. Baker, R. L. Waterfield, Division of Licensing and Regulation, U. S. Atomic Energy Commission, March 23, 1962.

4. Reg. Guide 1.25 Rev. 0 "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," Revision 0, March 23, 1972.

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6. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors," Baker, D. A., et. al., February 1988.

7. ANSI/ANS-18.1-1984, "American National Standard Radioactive Source Term Normal Operation of Light Water Reactors," American Nuclear Society, LaGrange Park, Illinois, Approved December 31, 1984.

8. ANS-18.1/ANSI N237-1976, "American National Standard Source Term Specification," American Nuclear Society, LaGrange Park, Illinois, Approved May 11, 1976.

9. NUREG-0017, Revision 1, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," April 1985.

8.0 OTHER LICENSING DOCUMENTATION

The Westinghouse scope for the Licensing Documentation deliverables for the HNP SGR/Uprating Analysis and Licensing Program included preparation of the following:

- Safety Evaluation (10CFR50.59)
- No Significant Hazards Evaluation (10CFR50.92)
- Proposed HNP Technical Specification and Bases Mark-ups
- Proposed HNP Final Safety Analysis Report Mark-ups

Westinghouse has completed these efforts and provided the documentation to CP&L under separate cover.

8.1 Safety Evaluation (10CFR50.59)

The final Safety Evaluation was provided to CP&L in Reference 1.

8.2 No Significant Hazards Evaluation (10CFR50.92)

The final No Significant Hazards Consideration Evaluation was provided to CP&L in Reference 2.

8.3 Proposed HNP Technical Specification and Bases Mark-ups

The final recommended Tech Spec and Bases mark-ups were provided to CP&L in Reference 3. Reference 4 provided an update to the recommended Tech Spec information and listing. A list of the mark-ups provided in Reference 4 is reproduced in Table 8-1 for information. Note that any subsequent changes to the Reference 4 list will supercede Table 8-1. The table provides the listing of the Tech Spec and Bases mark-ups and a reference to the Licensing Report Section that generated the mark-up.

The Tech Spec and Bases entries of Table 8-1 that are shown in **BOLD** are changes that CP&L may want to consider in the upcoming License Amendment Request (LAR). These changes were not addressed by Westinghouse in the SGR/Uprating licensing documentation (Westinghouse scope 10CFR50.92).

8.4 Proposed HNP Final Safety Analysis Report Markups

The final recommended FSAR mark-ups were provided to CP&L in Reference 5. Reference 6 provided an update to the recommended FSAR information and listing. A list of the mark-ups provided in Reference 6 is reproduced in Table 8-2 for information. Note that any subsequent changes to the Reference 6 list will supercede Table 8-2. The table provides the listing of the

FSAR mark-ups and a reference to the Licensing Report Section or related document that generated the mark-up.

8.5 References

1. CQL-00-270, Licensing Documentation – Final Safety Evaluation, SECL-00-043 Rev. 0.

2. CQL-00-269, Licensing Documentation – Final No Significant Hazards Consideration Evaluation, SECL-00-044 (SHC) Rev. 0.

3. CQL-00-234, Licensing Documentation – Final Recommended Technical Specification Mark-ups, June 20, 2000.

4. CQL-00-263, Licensing Documentation – Update No. 1 to Final Recommended Technical Specification Mark-ups, July 31, 2000.

5. CQL-00-227, Licensing Documentation – Final Recommended FSAR Mark-ups, June 13, 2000.

6. CQL-00-264, Licensing Documentation – Update No. 1 to Final Recommended FSAR Mark-ups, August 1, 2000.

Table 8-1
HNP SGR/Uprating Project
Recommended Tech Spec and Bases Updates

Licensing Report Section	T/S Section	T/S Page	Reason
1.0 Introduction	Definitions - 1.28 Rated Thermal Power	1-5	Revised/uprated licensed core power
2.0 NSSS Parameters			
2.1 Design PCWG Parameters	None	None	None
3.0 Design Transients			
3.1 NSSS Design Transients	**Table 5.7-1**	**5-8**	Suggested relocation of table.
3.2 Auxiliary Equipment Transients	None	None	None
4.0 NSSS Systems			
4.1 NSSS Fluid Systems	5.4.2 **Bases 3/4.4.2**	5-6 **B3/4 4-1**	Revised system volume due to RSG Update to safety valve relieving capacity (CP&L/AE/BOP).
4.2 NSSS/BOP Systems Interface	**3.7.1.3** **Bases 3/4.7.1.3** **Bases 3/4.1.2** Bases 3/4.7.1.1	**3/4 7-6** **B3/4 7-2** **B3/4 1-2a** B3/4 7-1	Change to Minimum CST usable volume. Update to volumes for BAT and RWST. (CP&L/AE/BOP). Update to steam line safety valve information based on revised analysis. Correction to delete discussion of "relieving capacities."

Table 8-1 (Cont.)
HNP SGR/Uprating Project
Recommended Tech Spec and Bases Updates

Licensing Report Section	T/S Section	T/S Page	Reason
4.3 NSSS Control Systems	**4.3.1.2** **Bases 3/4.3.1 & 3/4.3.2** **3.4.9.4b (Footnote)** **Figure 3.4-4** **Bases 3/4.4.9**	**3/4 3-1** **B3/4 3-2** **3/4 4-40** **3/4 4-41** **B3/4 4-14**	CP&L scope for completion of the NSAL-98-011 and LTOP setpoint analysis.
5.0 NSSS Components			
5.1.1 RV structural	**Bases Table B3/4.4-1**	**B3/4 4-8**	Update table to make it agree with FSAR Table 5.3.1-7. (CP&L Scope)
5.1.2 RV Integrity	**Not W Scope** - N/A	N/A	N/A
5.1.3 Incore Instrumentation	None	None	None
5.2 Reactor internals	None	None	None
5.3 Fuel Assemblies	**Not W Scope** - N/A	N/A	N/A
5.4 CRDMs	None	None	None
5.5.1 RCLP & Supports Nozzle Load Eval. Prim Comp Supports	None	None	None
5.5.2 RCLP LBB	None	None	None
5.5.3 Class 1 Aux. Lines	None	None	None

Table 8-1 (Cont.)
HNP SGR/Uprating Project
Recommended Tech Spec and Bases Updates

Licensing Report Section	T/S Section	T/S Page	Reason
5.6.1 RCP Structural	None	None	None
5.6.2 RCP Motors	None	None	None
5.7 Steam Generators 5.7.1 SG Structural-T/H 5.7.2 SG Thermal Hydraulic 5.7.3 Flow Induced Vibration and Wear	Index – 3/4.4 - RCS 4.4.5.2 4.4.5.3 4.4.5.4 4.4.5.5 Table 4.4-2A Table 4.4-2B Table 4.4-2C Bases - 3/4.4.5 3/4.4.7 Table 3.4-2 Table 4.4-3 Bases 3/4.4.7 Table 5.7-1	vii 3/4 4-13, -14 3/4 4-15 3/4 4-16, -16a 3/4 4-17 3/4 4-19 3/4 4-20 3/4 4-20a B3/4 4-3 3/4 4-26 3/4 4-27 3/4 4-28 B3/4 4-4, -5 5-8	RSG design eliminates FW pre-heaters. RSG tube selection, inspection and acceptance criterion and reporting requirements are changed. RSG eliminates F* criterion. Suggested relocation of the RSG revised chemistry limits due to added information from EPRI guideline TR-105714 (latest revision). Minor editorial update to secondary coolant system hydrostatic pressure.
5.8 Pressurizer	None	None	None
5.9 NSSS Auxiliary Equipment	None	None	None
6.0 Accident Analysis			
6.0.1 ICU	None	None	None

Table 8-1 (Cont.)
HNP SGR/Uprating Project
Recommended Tech Spec and Bases Updates

Licensing Report Section	T/S Section	T/S Page	Reason
6.1.1 6.1.2-SB/LB LOCA	**Not W Scope** - N/A	N/A	N/A
6.1.3 Post LOCA LTCC	None	None	None
6.1.4 HLSO	None	None	None
6.1.5 Rod Ejection Accident	**Not W Scope** - N/A	N/A	N/A
6.2 Non-LOCA	**Not W Scope** - N/A	N/A	NA/
6.3.1 SGTR T/H - Overfill	None	None	None
6.3.2 SGTR T/H - Dose	None	None	None
6.3.3 SGTR Radiological Evaluation	Definitions - 1.11 – Dose Equivalent I-131 3.4.8 Figure 3.4-1 Table 4.4-4 Bases 3/4.4.8	1-2a 3/4 4-29 3/4 4-30 3/4 4-31 B3/4 4-5	Update to reflect the latest radiological analysis for the SGR/Uprate which used dose conversion factors of ICRP-30 and 0.35μCi/g
6.4 LOCA LT/ST M&E	None	None	None
6.5.1 MSLB M&E IC	None	None	None
6.5.2 MSLB M&E OC	None	None	None
6.5.3 MSLB M&E Radiological	None	None	None
6.6 LOCA Forces	None	None	None

Table 8-1 (Cont.) HNP SGR/Uprating Project Recommended Tech Spec and Bases Updates			
Licensing Report Section	**T/S Section**	**T/S Page**	**Reason**
6.7 RTS/ESFAS Setpoints	2.2 Table 2.2-1 Bases 2.2.1 3.3.2 Table 3.3-4 Bases 3/4.3.1 and 3/4.3.2	2-1, -3 2-4 Thru –10 B2-2, -3 3/4 3-16 3/4 3-28 Thru –35 B3/4 3-1, 3-2	Initial condition uncertainty and RTS/ESFAS setpoint changes.
	3.4.1.2 4.4.1.2.2 4.4.1.3.2 3.4.1.4.1.b 4.4.1.4.1.1 3.10.4 Bases 3/4.4.1	3/4 4-2 3/4 4-3 3/4 4-5 3/4 4-6 3/4 10-4 B3/4 4-1	Recalculation of the NR/WR level for SG heat sink. (This work was not related to RTS/ESFAS but was a separate work effort.) Also related to SG level is the Special Test Exception.
6.8 ATWS	None	None	None
6.9 Natural Circulation Cooldown	None	None	None
7.0 Nuclear Fuel			
7.1, 7.2, 7.3 CD, T-H, FR Des	**Not W Scope** - N/A	N/A	N/A
7.4 Heat Generation Rates	None	None	None
7.5 Neutron Fluence	**Not W Scope** - N/A	N/A	N/A

Table 8-1 (Cont.) HNP SGR/Uprating Project Recommended Tech Spec and Bases Updates			
Licensing Report Section	**T/S Section**	**T/S Page**	**Reason**
7.6 Source Terms	None	None	None
8.0 No Impact Systems			
8.1 Mechanical Systems	None	None	None
8.2 I&C Systems	None	None	None

Table 8-2
HNP SGR/Uprating Project
Recommended FSAR Updates

Licensing Report Section	FSAR Pages Section/Tables /Figures	FSAR Pages	Reason
1.0 Introduction	1.1.5	1.1-2	Change in the licensed core and NSSS power
	Table 1.1.1-1 Table 1.1.1-2	1.1-7, -10, -11 1.1-13, -14, -15	Updates to show changes due to SGR/Uprating work
	1.6	1.6-2, -4, -5, -10, -13	Updates to include the methodologies used in the SGR/Uprating project
2.0 NSSS Parameters			
2.1 Design PCWG Parameters	5.1.0 Table 5.1.0-1	5.1.0-3, -4 5.1.0-6, -7	General NSSS parameter changes (e.g., power, Tavg, SGTP, FW temperatures).
3.0 Design Transients			
3.1 NSSS Design Transients	3.9.1.1.1 3.9.1.1.2 3.9.1.1.4 3.9.1.1.5 Table 3.9.1-1	3.9.1-2, -3, -4 3.9.1-5, -6, -7 3.9.1-8, -8a 3.9.1-8a, -9 3.9.1-27, -28	Update to reflect the analysis condition transients
3.2 Auxiliary Equipment Design Transients	None	None	None

Table 8-2 (Cont.) HNP SGR/Uprating Project Recommended FSAR Updates			
Licensing Report Section	**FSAR Pages Section/Tables /Figures**	**FSAR Pages**	**Reason**
4.0 NSSS Systems			
4.1 NSSS Fluid Systems	Table 5.1.0-1	5.1.0-6, -7	General parameter changes (e.g., power, Tavg, SGTP, FW temperatures).
	Figure 5.1.1-1 Table 5.4.13-1	1 (Notes) 5.4.13-3	General parameter changes
	5.4.7.1 5.4.7.3 Table 5.4.7-1	5.4-7-1 5.4.7-10j 5.4.7-11	RHR Design bases/performance evaluation revisions.
	6.3.2.8 Table 6.3.2-8 Table 6.3.2-9	6.3.2-13d, -e, -f 6.3.2-28 6.3.2-29, -30	General parameter changes. To be confirmed by CP&L/Raytheon.
	9.2.1.3.1 Table 9.2.1-4 Table 9.2.1-7 Table 9.2.1-8 Table 9.2.1-9	9.2.1-6, -7 9.2.1-19 9.2.1-22 9.2.1-23 9.2.1-24	Revised RHR cool down analysis.
	9.2.2.1	9.2.2-1	

Table 8-2 (Cont.) HNP SGR/Uprating Project Recommended FSAR Updates			
Licensing Report Section	**FSAR Pages Section/Tables /Figures**	**FSAR Pages**	**Reason**
	9.2.2.2.1 9.2.2.2.3 9.2.2.3 Table 9.2.2-1 Table 9.2.2-3	9.2.2-3 9.2.2-6, -7 9.2.2-8 9.2.2-10 9.2.2-12	General parameter updates. CCW pump data to be confirmed/revised/specified by Raytheon.
	9.3.4.1.2.5 9.3.4.1.2.6 9.3.4.2.2	9.3.4-14, -17, -20 9.3.4-23, -24 9.3.4-32	General updates
	Table 9.3.4-1	9.3.4-40	General parameter changes
4.2 NSSS/BOP Fluid System Interface	9.2.6.3 Table 10.1.0-1 Table 10.3.1-1	9.2.6-2 10.1.0-4 10.3.1-5, -6	Change to Minimum CST usable volume (AE/BOP) Changes to the NSSS parameters Changes based on NSSS/BOP requirements in SSDC Rev. 3 and Δ75 T-H report
	10.4.4 Table 10.4.4-1 10.4.7 Table 10.4.9-1	10.4.4-2 10.4.4-4 10.4.7-1,-2,-4 10.4.9-7	Changes based on latest version of the W equipment specifications, G-67840 Rev. 3 and 952228 Rev. 4 Changes based on Δ75 design (W SG Design) Change to Minimum CST usable volume (AE/BOP)

Table 8-2 (Cont.) HNP SGR/Uprating Project Recommended FSAR Updates			
Licensing Report Section	**FSAR Pages Section/Tables /Figures**	**FSAR Pages**	**Reason**
4.3 NSSS Control Systems	5.2.2 Table 5.4.13-1 Table 10.3.1-1 7.7.1 7.7.1.7 7.7.1.8 7.7.2.5 Table 15.0.6-2 Table 16.3-1	5.2.2-1 5.4.13-3 10.3.1-5 7.7.1-3 7.7.1-11 7.7.1-12 7.7.2-6 15.0.6-4 16.3-2	Update to incorporate the latest analysis and general corrections.
5.0 NSSS Components			
5.1.1 RV structural	Table 5.2.1-1 5.2.1.1 5.3.1.2	5.2.1-3 5.2.1-1 5.3.1-1	Update to correct the ASME code reference Update to clarify RV code case compliance and Appendix G evaluation and limiting SG RT_{NDT}.
5.1.2 RV Integrity	Not W Scope	N/A	N/A
5.1.3 Incore Instrumentation	None	None	None
5.2 Reactor internals	3.7.3.14 3.7 References	3.7.3-19	General update and clarification of W fuel
5.3 Fuel Assemblies	Not W Scope	N/A	N/A
5.4 CRDMs	None	None	None

Table 8-2 (Cont.) HNP SGR/Uprating Project Recommended FSAR Updates			
Licensing Report Section	**FSAR Pages Section/Tables /Figures**	**FSAR Pages**	**Reason**
5.5.1 RCLP & Supports Nozzle Load Evaluation Primary Component Supports	3.6.2.1.1.1 3.6.2.5.1 Table 3.6.2-1 Table 3.6.2-2 Figure 3.6.2-9 3.9.1.4.3 Table 3.9.1-1 Table 3.9.1-5 Figure 5.4.14-1 Figure 5.4.14-3	3.6.2-1 3.6.2-18 3.6.2-19 3.2.6-20 -- 3.8.1-16, -17 3.9.1-28 3.9.1-32 -- --	Update to incorporate the latest analysis, LBB applicability and general corrections.
5.5.2 RCLP LBB	3.6 3.6.2.1.1.1, 3.6.2.2.1 Table 3.6.2-2, References 3.6	3.6.0-2 3.6.2-1 3.6.2-8, -8a, -9, -10 3.6.2-20 --	Incorporation of the latest LBB work, WCAP 14549 and general updating for the elimination of large breaks via LBB implementation. Elimination of historical discussion regarding large primary system breaks.
5.5.3 Class 1 Auxiliary Lines	Figure 3.6A-20.1 CALC, Shts 1, 2 & 3 Figure 3.6A-23 CALC Figure 3.6A-PLOT-A, -B & -C	- - -	Update to reflect the latest piping analysis

Table 8-2 (Cont.) HNP SGR/Uprating Project Recommended FSAR Updates			
Licensing Report Section	**FSAR Pages Section/Tables /Figures**	**FSAR Pages**	**Reason**
5.6.1 RCP Structural	None	None	None
5.6.2 RCP Motors	Table 5.4.1-1	5.4.1-10	Revised RCP parameters due to PCWG and BEF changes
5.7 Steam Generators	1.2.2.2	1.2.2-2	RSG design has different MCO value.
5.7.1 SG Structural-T/H	Table 1.3.2-1	1.3.2-2	RSG design eliminates FW pre-heaters.
5.7.2 SG Thermal Hydraulic	Reg. Guide 1.3.4	1.8-46	RSG uses different construction standards
5.7.3 Flow Induced	5.2.1.2	5.2.1-2	Code case changes and eliminates RG applicability.
Vibration and Wear	Table 5.2.1-1	5.2.1-3	RSG has different ASME Code Addenda.
	Table 5.2.1-2	5.2.1-5	RSG has different code case applicability.
	5.2.3.1	5.2.3-1, -4, -5	RSG has different materials, specifications and material
	Table 5.2.3-1	5.2.3-12, -13	properties.
	5.4.2	5.4.2-1 - 7a	
	Table 5.4.2-1	5.4.2-8	RSG necessitates total rewrite since the design is
	Table 5.4.2-2	5.4.2-9, -10	different than original steam generators.
	Figure 5.4.2-1		
	5.4.4.2	5.4.4-1	
	Table 5.7-1	5-8	RSG design has different steam flow restrictor design.
	Table 10.1.0-1	10.1.0-4	RSG manufacturing performed hydro test prior to ship. Changes based on system parameter changes.
	7.1.2.5	7.1.2-11, -12	

Table 8-2 (Cont.) HNP SGR/Uprating Project Recommended FSAR Updates			
Licensing Report Section	**FSAR Pages Section/Tables /Figures**	**FSAR Pages**	**Reason**
	7.3.2.2.10.7 Table 7.3.1-11 Figure 7.3.1-8 Sheet 1, 2 and 3 10.3.5.2 10.4.7 10.4.9.2.4 Table 12.3.1-1 Table 12.3.1-2 15.6.3.1	7.3.2-8 7.3.1-68 -- 10.3.5-1 10.4.7-1. -2, -4, -6 10.4.9-4 12.3.1-10 12.3.1.-11 15.6.3-1a	Elimination of steam generator preheater wording due the Δ75 design. RSG chemistry general update RSG Model Δ75 design eliminates FW pre-heaters and FW system changes. General update RSG has different design for wetted surface area. RSG has different tube material.
5.8 Pressurizer	None	None	None
5.9 NSSS Auxiliary Equipment	None	None	None
6.0 Accident Analysis			
6.0.1 Initial Condition Uncertainties	None	None	None

Table 8-2 (Cont.) HNP SGR/Uprating Project Recommended FSAR Updates			
Licensing Report Section	**FSAR Pages Section/Tables /Figures**	**FSAR Pages**	**Reason**
6.1.1, 6.1.2-SB/LB LOCA	Not W Scope	N/A	N/A
6.1.3 Post LOCA LTCC	None	None	None
6.1.4 HLSO	None	None	None
6.1.5 Rod Ejection Accident	Not W Scope	N/A	N/A
6.2 Non-LOCA	Not W Scope	N/A	N/A
6.3.1 SGTR Margin to Over Fill	15.6.3 Reference Sect 15.6 Table 15.6.3-1 Table 15.6.3-2 Figures 15.6.3-1 Through Figure 15.6.3-6	15.6.3-1, 1a, 2, 4, 4a, 4b, 4c, 4d 15.6R-1 15.6.3-13 15.6.3-14 --	Update to reflect the revised T/H overfill analysis

Table 8-2 (Cont.) HNP SGR/Uprating Project Recommended FSAR Updates			
Licensing Report Section	**FSAR Pages Section/Tables /Figures**	**FSAR Pages**	**Reason**
6.3.2 SGTR T/H - Doses	15.6.3.3.1 15.6.3.3.2 15.6.3.3.3 15.6.3.3.4 Table 15.6.3-3 Table 15.6.3-4 Table 15.6.3-5 Figure 15.6.3-7 Through Figure 15.6.3-17 and Figure 15.6.3-19	15.6.3-5 15.6.3-6 15.6.3-6, -7, -7a 15.6.3-8 15.6.3-15 15.6.3-16 15.6.3-17 --	Update to reflect the revised T/H analysis for doses
6.3.3 SGTR Radiological Evaluation	15.6.3.4 Tables 15.6.3-6 Through 15.6.3-13 Reference Sect 15.6 Figure 15.6.3-18 Figure 15.6.3-20 Figure 15.6.3-21	15.6.3-9, -10, -11, 12, 12a 15.6.3-18 Through 15.6.3-26 15.6R-1 - - -	Update to reflect the revised dose analysis

Table 8-2 (Cont.) HNP SGR/Uprating Project Recommended FSAR Updates			
Licensing Report Section	**FSAR Pages Section/Tables /Figures**	**FSAR Pages**	**Reason**
6.4 LOCA LT/ST M&E	6.2.1.1.3.2 6.2.1.2a Table 6.2.1-17 Table 6.2.1-18 6.2.1.3 Table 6.2.1-5 References 6.2 Table 6.2.1-28 Through Table 6.2.1-56	6.2.1-5 6.2.1-19a, -19b 6.2.1-90 6.2.1-91 6.2.1-20 – 24 6.2.1-35 6.2R-1, -2 6.2.1-113 Through 161	Power uprate, revised PCWG and M&E releases

Table 8-2 (Cont.) HNP SGR/Uprating Project Recommended FSAR Updates			
Licensing Report Section	**FSAR Pages Section/Tables /Figures**	**FSAR Pages**	**Reason**
6.5.1 MSLB M&E IC	Table 1.6-1 3.6A.3.2 3.11.4.5 3.11.6 App. 3.11C App. 3.11E 6.2.1.1.1 6.2.1.1.3 6.2.1.1.3.2 6.2.1.4 Table 6.2.1-1 Table 6.2.1-2 Table 6.2.1-3 Table 6.2.1-4 Table 6.2.1-5 Table 6.2.1-9 Table 6.2.1-58A Reference App. 6.2B App. 6.2C	1.6-4, -10 3.6A-14, -14a 3.11.4-2 3.11.6-1 3.11C-1 3.11E-1, -2, -4 6.2.1-1, –2 6.2.1-4, -5 6.2.1-5, -9, -11, -12 6.2.1-24, 26a 6.2.1-27, -28 6.2.1-30 6.2.1-31 6.2.1-32 6.2.1-35 6.2.1-49, -50 6.2.1-163, -163a 6.2R-2 6.2B-1 6.2C-1	General update to include the latest analysis methodology Power uprate, revised SG MCO and M&E releases
6.5.2 MSLB M&E OC	See Section 6.5.1, above for changes	See Section 6.5.1	Power uprate, revised SG MCO and M&E releases

Table 8-2 (Cont.) HNP SGR/Uprating Project Recommended FSAR Updates			
Licensing Report Section	**FSAR Pages Section/Tables /Figures**	**FSAR Pages**	**Reason**
6.5.3 MSLB M&E Radiological	None	None	None
6.6 LOCA Forces	None 3.9.1.4.3 3.9.1.4.6	None 3.9.1-17 3.9.1-21 Through -25	None due to SGR/Uprating 1994 through 1996 Harris T-Hot Reduction, Increased SGTP program mark-ups that were not incorporated by CP&L. FSAR mark-ups are not part of the current SGR/Uprating effort they are still valid and were reissued for CP&L to incorporate.
6.7 RTS ESFAS Setpoints	7.2.1.1.2 7.2.1.2.4 Table 7.2.1-3 References 7.2 Table 7.3.1-11 Table 7.3.1-12 7.7 References 7.7 Figure 7.7.1-6 Table 15.0.6-2	7.2.1-4, -5, -6 7.2.1-13 7.2.1-19, 20 7.2R-1 7.3.1-37 7.3.1-70 7.7.1-1, -2 7.7R-1 -- 15.0.6-3, -4, -5	Initial condition uncertainty and RTS/ESFAS setpoint changes
6.8 ATWS	15.8	15.8.1-1, -2	Incorporate the revised RSG/Uprate conditions

Table 8-2 (Cont.) HNP SGR/Uprating Project Recommended FSAR Updates			
Licensing Report Section	**FSAR Pages Section/Tables /Figures**	**FSAR Pages**	**Reason**
6.9 Natural Circulation Cooldown	1.8 5.4.7.2.8	1.8-177, -177a 5.4.7-10e, f, g, h, i, j & k	Revised natural circulation analysis for compliance with RSB BTP 5-1
7.0 Nuclear Fuel			
7.1, 7.2, 7.3 CD, T-H, FR Design	Not W Scope	N/A	N/A
7.4 Heat Generation Rates	None	None	None
7.5 Neutron Fluence	Not W Scope	N/A	N/A
7.6 Source Terms	11.1.1 11.1.1.1 11.1.1.2 Table 11.1.1-1 Table 11.1.1-2 Table 11.1.1.-3 Table 11.1.1-4 Table 11.1.1-5 11.1.2 Table 11.1.2-1 Table 11.1.2-2 11.3.1	11.1.1-1 11.1.1-1, -3 11.1.1-3, -4 11.1.1-5, -6, -7, -8, -9, -10 11.1.1-11, -12 11.1.1-13, -14 11.1.1-15, -16 11.1.1-17, -18 11.1.2-1 11.1.2-2, -3 11.1.2-4 11.3.1-2	Revised source terms from power uprate

Table 8-2 (Cont.) HNP SGR/Uprating Project Recommended FSAR Updates			
Licensing Report Section	**FSAR Pages Section/Tables /Figures**	**FSAR Pages**	**Reason**
	Table 11.3.2-1	11.3.2-10	
	Table 11.3.2-2	11.3.2-11	
	Table 11.3.2-3	11.3.2-12	
	11.1.4	11.1.4-1	
	Table 11.1.4-1	11.1.4-2	
	Table 11.4.1-2	11.4.1-4, -5	
	Table 11.4.1-3	11.4.1-6,-7	
	Table 11.4.1-4	11.4.1-8, -9	
	Table 11.4.1-5	11.4.1-10, -11	
	Table 11.1.6-1	11.1.6-4	
	15.0.9	15.0.9-1	
	Table 15.0.9-1	15.0.9-3	
	References 15.0	15.0R-1	
	15.6.3.4.3	15.6.3-10	
	Table 15.6.3-9	15.6.3-22	
	Table 15.6.5-14	15.6.5-34, -35	
	15.7.1.2	15.7.1-1	
	15.7.1.3.1	15.7.1-2	
	Table 15.7.1-1	15.7.1-3	
	Table 15.7.1-2	15.7.1-4	
	Table 15.7.2-2	15.7.2-3	
	Table 15.7.2-3	15.7.2-4	
	15.7.4.1	15.7.4-1	

Table 8-2 (Cont.) HNP SGR/Uprating Project Recommended FSAR Updates			
Licensing Report Section	**FSAR Pages Section/Tables /Figures**	**FSAR Pages**	**Reason**
	15.7.4.2.1 15.7.4.2.2 Table 15.7.4-1 Table 15.7.4-3 Table 15.7.4-8 References 15.7	15.7.4-1, -2 15.7.4-2 15.7.4-8 15.7.4-11 15.7.4-18 15.7R-1	
W Radiation Manual (PCN #5) (Not in the LR but is a related SGR/Uprating document)	12.2.1 Table 12.2.1-1, -2 ,-3 Table 12.2.1-9 Through -21 Table 12.2.1-23, -24 References 12.2 Figures 12.2.1-1, -2, -3	12.2.1-1 Through –4 12.2.1-5, -6, -7 12.2.1-13 Through –25 12.2.1-27, -28 --	These revisions were made based on W review of the FSAR Section 12.2 as described in PCN#5. This information was issued previously as preliminary via CQL-00-241. This information is now finalized and there are no changes from the preliminary information.

Proprietary Information Notice For NRC Transmittal Letter

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10CFR2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) contained within parentheses located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10CFR2.790(b)(1).

Copyright Notice For NRC Transmittal Letter

The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.790 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

Westinghouse Electric Company, LLC

Box 355
Pittsburgh Pennsylvania 15230-0355

August 31, 2000

CAW-00-1415

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. Samuel J. Collins

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject "Shearon Harris Nuclear Power Plant, Steam Generator Replacement/Uprate Analysis and Licensing Project, NSSS Licensing Report WCAP-15398" [Proprietary]

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-00-1415 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Carolina Power and Light Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-00-1415 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

Cc: S. Bloom/NRR/OWFN/DRPW/PDIV2 (Rockville, MD) 1L

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CAW-00-1415

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Henry A. Sepp, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Henry A. Sepp, Manager

Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 31st day
of August, 2000

Lorraine M. Piplica

Notary Public

Notarial Seal
Lorraine M. Piplica, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Dec. 14, 2003
Member, Pennsylvania Association of Notaries

0535s.doc

(1) I am Manager, Regulatory and Licensing Engineering, in the Nuclear Services Business Unit of the Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Electric Company LLC in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Shearon Harris Nuclear Power Plant, Steam Generator Replacement/Uprate Analysis and Licensing Project, NSSS Licensing Report WCAP-15398" [Proprietary] being transmitted by Carolina Power and Light Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk, Attention Mr. Samuel J. Collins. The proprietary information as submitted for use by Carolina Power and Light for the Shearon Harris Nuclear Plant is expected to be applicable in other licensee submittals in response to certain NRC requirements for steam generator replacement and uprating.

This information is part of that which will enable Westinghouse to:

(a) Provide documentation of the methods for determining acceptable plant operation at the Steam Generator Replacement/Uprate conditions.

(b) Provide the specific analysis or evaluation results related to the parameters that are considered for the Steam Generator Replacement/Uprating project.

(c) Assist the customer to obtain NRC approval.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of the technology to its customers in the licensing process.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculation, evaluation and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing analytical methods and performing tests.

Further the deponent sayeth not.