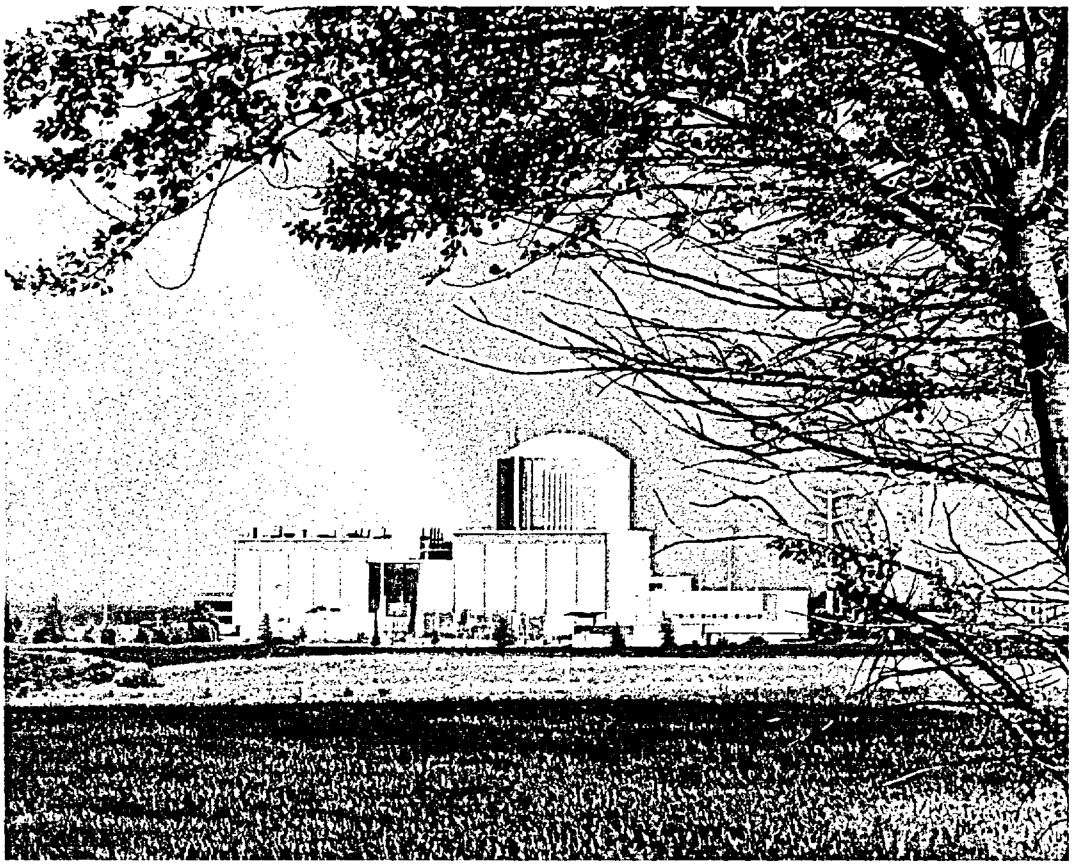
Westinghouse Non-Proprietary Class 3

WCAP-16040-NP

February 2003

Power Uprate Project

Kewaunee Nuclear Power Plant



NSSS and BOP Licensing Report

Westinghouse

Stone & Webster™
A Shaw Group Company

Westinghouse Non-Proprietary Class 3

WCAP-16040-NP

February 2003

Power Uprate Project
Kewaunee Nuclear Power Plant

NSSS and BOP Licensing Report

Westinghouse Non-Proprietary Class 3

WCAP-16040-NP

Power Uprate Project
Kewaunee Nuclear Power Plant

NSSS and BOP Licensing Report

R. H. Owoc
Westinghouse Nuclear Services

J. R. Stukus
Westinghouse Nuclear Services

February 28, 2003

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

6080\frontmatter.doc-022703

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont.)

TABLE OF CONTENTS (Cont.)

TABLE OF CONTENTS (Cont.)

TABLE OF CONTENTS (Cont.)

TABLE OF CONTENTS (Cont.)

TABLE OF CONTENTS (Cont.)

TABLE OF CONTENTS (Cont.)

LIST OF TABLES

LIST OF TABLES (Cont.)

LIST OF TABLES (Cont.)

LIST OF TABLES (Cont.)

LIST OF TABLES (Cont.)

LIST OF TABLES (Cont.)

LIST OF FIGURES

LIST OF FIGURES (Cont.)

LIST OF ACRONYMS

3-D	three-dimensional
ACSR	Aluminum Cable Steel Reinforced
AFS	Auxiliary Feedwater System
AFW	auxiliary feedwater
AIF	Atomic Industrial Forum
AISC	American Institute of Steel Construction
ALARA	as-low-as-is-reasonably-achievable
AMSAC	ATWS mitigating system actuation circuitry
ART	adjusted reference temperature
ARV	atmospheric relief valve
ASDV	atmospheric steam dump valve
ASN	Analysis Section Number
AST	alternative source term
ATWS	anticipated transient without scram
AVB	anti-vibration bar
B&PV	boiler and pressure vessel
BELBLOCA	best-estimate large-break loss-of-coolant accident
BELOCA	best-estimate LOCA
BE-UPI	best-estimate upper plenum injection
BOC	beginning of cycle
BOL	beginning of life
BOP	balance of plant
C&FS	Condensate and Feedwater System
CCS	Containment Air Cooling System
CCWS	Component Cooling Water System
CDV	condenser dump valve
CEDE	committed effective dose equivalent
CFCU	containment fan coil unit
CFD	computational fluid dynamics

LIST OF ACRONYMS (Cont.)

CFR	Code of Federal Regulations
CN	calculation note
COLR	Core Operating Limit Report
CRDM	control rod drive mechanism
CS	containment spray
CSA	channel statistical allowance
CSS	Containment Spray System
CST	condensate storage tank
CT	current transformer
Cv	valve flow coefficient
CVCS	Chemical and Volume Control System
CT	current transformer
CW	circulating water
CWIT	circulating water inlet temperature
CWS	Circulating Water System
DBA	design basis accident
DCF	dose conversion factor
DE	dose equivalent
DEHL	double-ended hot leg
DEPS	double-ended pump suction
dpa	displacement of atom
DSS	Diverse Scram System
EBOP	emergency bearing oil pump
ECCS	Emergency Core Cooling System
EDG	emergency diesel generator
EFPD	effective full-power day
EFPY	effective full-power year
EM	evaluation model
EOC	end of cycle

LIST OF ACRONYMS (Cont.)

EOL	end of license
EOLE	end-of-life extension
EPA	Environmental Protection Agency
EQ	environmental qualification
ERG	Emergency Response Guideline
ESDR	Engineered Safeguards Design Rating
ESF	emergency safeguard feature
ESOP	emergency seal oil pump
FAC	flow-accelerated corrosion
FBCV	feedwater bypass control valve
FCV	fan cooling unit
FHA	fuel-handling accident
FIV	flow-induced vibration
FPS	full-power second
FRV	feedwater control valve
FSAR	Final Safety Analysis Report
FU	fuel upgrade
FWRV	feedwater regulation valve
GDC	General Design Criteria
GDT	gas decay tank
HEI	Heat Exchange Institute, Inc.
HELB	high-energy line break
HFF	hydraulic forcing function
HHSI	high-head safety injection
HVAC	heating, ventilation, and air conditioning
ID	inside diameter
IPB	Iso-Phase bus
KNPP	Kewaunee nuclear power plant
LBB	leak before break

LIST OF ACRONYMS (Cont.)

LBLOCA	large-break loss-of-coolant accident
LHF	LOCA hydraulic force
LHSI	low-head safety injection
LOCA	loss-of-coolant accident
LONF	loss of normal feedwater
LPZ	low-population zone
LTCC	long-term core cooling
LTOP	low-pressure overpressure protection
LWR	light-water reactor
MAT	main auxiliary transformer
M/G	motor/generator
MAT	main auxiliary transformer
MCC	Motor Control Center
MCO	moisture carryover
MDLM	mist diffusion layer model
MMF	minimum measured flow
MOC	middle of cycle
MSIV	main steam isolation valve
MSLB	main steamline break
MS	main steam
MSS	Main Steam System
MSSV	main steam safety valve
MT	main transformer
MTC	moderator temperature coefficient
MTU	metric ton unit
NAI	Numerical Applications Incorporated
NDE	nondestructive examination
NEC	National Electric Code
NEMA	National Electric Manufacturer's Association

LIST OF ACRONYMS (Cont.)

NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NRS	narrow range span
NSSS	Nuclear Steam Supply System
NUPPSCO	Nuclear Power Plant Standards Committee
OBE	operating basis earthquake
OD	outside diameter
ODSCC	outer diameter stress corrosion cracking
OEM	Original Equipment Manufacturer
OPΔT	overpressure ΔT
ORNL	Oak Ridge National Laboratory
OTΔT	overtemperature ΔT
PBNP	Point Beach Nuclear Plant
PCWG	Performance Capability Working Group
PORV	power-operated relief valve
PRT	pressurizer relief tank
PSS	Primary Sampling System
PT	potential transformer
P-T	pressure-temperature
PTS	pressurized thermal shock
PU	power upgrade
PWR	pressurized-water reactor
PWSCC	primary water stress corrosion cracking
RAT	reserve auxiliary transformer
RCCA	rod control cluster assembly
RCL	reactor coolant loop
RCPON	reactor coolant pump outlet nozzle
RCS	Reactor Coolant System
RHR	residual heat removal

LIST OF ACRONYMS (Cont.)

RHRS	Residual Heat Removal System
RPV	reactor pressure vessel
RPVIN	reactor pressure inlet nozzle
RSAC	reload safety analysis checklist
RSE	Reload Safety Evaluation
RSG	replacement steam generator
RTD	resistance temperature detector
RTDP	revised thermal design procedure
RTP	rated thermal power
RT_{PTS}	reference temperature-pressurized thermal shock
RTSR	Reload Transition Safety Report
RWFS	rod withdrawal from subcritical
RWST	refueling water storage tank
SB	site boundary
SBV	Shield Building ventilation
SCC	stress corrosion cracking
scfm	standard cubic feet per minute
SER	Safety Evaluation Report
SF	scaling factor
SF	service factor
SG	steam generator
SGBS	Steam Generator Blowdown System
SGR	steam generator replacement
SGTP	steam generator tube plugging
SGTR	steam generator tube rupture
SI	safety injection
SIA	Structural Integrated Associates
SIS	Safety Injection System
SRP	Standard Review Plan

LIST OF ACRONYMS (Cont.)

SRSS	square root of the sum of the squares
SS	Sampling System
SSE	safe shutdown earthquake
STDP	standard thermal design procedure
SW	service water
SWPC	Siemens-Westinghouse Power Corporation
SWS	Service Water System
TA	total allowance
TAT	tertiary auxiliary transformer
T_{ave}	operating condition temperature
T_{avg}	average temperature[1]
T_{cold}	cold leg temperature
TDF	thermal design flow
TDH	total discharge head
TEDE	total effective dose equivalent
T_{feed}	feedwater temperature
T_{hot}	hot leg temperature
TID	total integrated dose
T_{ref}	reference temperature[1]
T_{sat}	water at pressurizer temperature or saturation temperature
TSF	temperature scaling factor
TSP	tube support plate
T_{steam}	steam temperature
UEF	unfavorable exposure time
UFM	ultrasonic flow meter
USE	upper shelf energy

[1] RCS or vessel temperature, depending on context.

LIST OF ACRONYMS (Cont.)

UTM	ultrasonic temperature measurement
v/o	volume percent
VCT	volume control tank
WDNR	state of Wisconsin, Department of Natural Resources

1.0 INTRODUCTION

1.1 Purpose

The purpose of performing the Kewaunee Nuclear Power Plant (KNPP) power uprate analyses and evaluations is to demonstrate that the Nuclear Steam Supply Systems (NSSSs) and Balance of Plant (BOP) will remain in compliance with applicable licensing criteria and requirements and operate acceptably at the increased thermal/electrical power conditions. This report summarizes the results of the analyses and evaluations.

1.2 Introduction and Background

KNPP was originally designed with equipment and systems capable of accommodating operating conditions above the original licensed power rating including higher pressures, flows, and temperatures. In addition, continuing improvements in the analytical techniques, measurement instrument accuracies, plant thermal performance, and fuel and core designs have resulted in an increased margin between the safety analyses results and the licensing limits. These available safety margins, combined with the excess margin in the as-designed equipment, system and component capabilities, provide KNPP with the potential for an increase in thermal power rating of 7.4 percent without major NSSS or BOP hardware modifications with no significant increase in the hazards presented by the plant as approved by the NRC at the original license stage.

The current KNPP operating license issued by the NRC is for a rated reactor core power of 1650 MWt and is based upon the design and accident analysis developed by Westinghouse Electric Corporation, now known as Westinghouse Electric Company LLC. It was, however, the standard practice of Westinghouse in performing these analyses to use Engineered Safeguards Design Rating (ESDR) of 104.5-percent rated power for some of the analyzed events. Therefore, the engineering safeguards equipment design and the original plant safety analyses supported a higher reactor power than the reactor power that was licensed. In 1996, a study was undertaken to evaluate the option of increasing the electrical output on KNPP. At that time, however, the capacity of KNPP was restricted due to degraded steam generator tubing. Therefore, it was concluded that a power uprate would require replacement of the steam generators. Efforts to increase the electrical capacity of KNPP were deferred until a decision was made to replace the steam generators.

With the steam generator program underway, NMC again entered into discussions with several vendors on a potential KNPP Uprate Project. Deliberations included development of a success path that utilized the analytical and licensing work being undertaken both for replacing the degraded steam generators and also transitioning to a new fuel product. Of significance in these efforts were the licensing of a new T_{avg} window for operation of KNPP, and the implementation of an alternate source term program. Additionally NMC also decided to implement a measurement uncertainty recapture (MUR) power uprate project prior to implementation of the stretch uprate.

Prior to initiation of the design capacity recovery or stretch uprate project, NMC commissioned a feasibility study in order to formalize the scope of these activities and develop a detailed project plan and schedule to uprate KNPP from 1650- to 1749-MWt rated core power, a 6-percent power uprate. A contract was awarded to Westinghouse Electric Company LLC to assist NMC in this effort. Westinghouse scope focused on the evaluation of the NSSS. Stone and Webster (S&W) and the Siemens-Westinghouse Power Corporation (SWPC) were subcontracted to assist in the evaluation of the secondary plant and the turbine-generator, respectively. The results of the feasibility study indicated that the current rated core power could be increased by approximately 6 percent without the need for major plant modifications.

Subsequent to the feasibility study, NMC also decided to incorporate an Appendix K or measurement uncertainty recapture power uprate with the use of a more accurate feedwater flow measurement device. This provided an additional 1.4-percent power uprate. Thus, the 7.4-percent uprate consists of a 6-percent design capacity recovery uprate (stretch uprate), reviewed in the feasibility study, and a 1.4-percent measurement uncertainty recapture uprate as a result of the reduced power measurement uncertainties.

It's important to note that the KNPP Uprate is organized in such a way to capitalize on the synergies associated with analyses performed for the Replacement Steam Generator (RSG) Project and the Fuel Transition Project. The plant operating temperature program for the Plant Uprate Project is designed to maintain the current minimum reactor vessel inlet temperature, T_{cold}, and maximum vessel outlet temperature, T_{hot}, as defined in the RSG project, thus clipping the T_{avg} operating window by approximately 2.5°F on either side. By maintaining the maximum T_{hot} and minimum T_{cold}, several of the NSSS systems and component analyses performed for the RSG and Fuel Transition Projects remain bounding and applicable to the Uprate Project. The Power Uprate Project's organization includes interdependencies with other projects, such as RSG and Fuel Transition, and requires that its licensing strategy be similarly organized for the overall uprate implementation. The engineering, analysis, and licensing strategy for KNPP power uprate minimized duplication of effort and subsequent review time on the part of NRC, hence enhancing NMC's ability to obtain approval of a stretch power uprate license amendment application.

1.3 Scope

In support of the KNPP uprate, the following organizations have performed analyses and evaluations to demonstrate that KNPP will remain in compliance with applicable licensing criteria and requirements at the uprated power levels.

- Nuclear Management Company
- Westinghouse Electric Company
- Stone & Webster Engineering Corporation
- Siemens-Westinghouse Power Corporation (SWPC)
- Advanced Measurement and Analysis Group, Inc. (AMAG)

The scope of the above organizations is as follows:

1.3.1 Nuclear Management Company

As, licensee, and operator, Nuclear Management Company (NMC) has the overall technical, contractual and commercial oversight and decision making responsibility for the KNPP Power Uprate. NMC, as Lead Project Manager, is responsible for oversight of the program, and has monitored the performance of its subcontractors and support organizations regarding scope of responsibility, quality of performance, compliance with schedules, and communication among team member organizations. NMC controlled the progress of the overall project with input from each of the team member organizations. NMC reviewed and authorized revisions to the project scope and schedule and managed the commercial implications of those changes. NMC was responsible for contract management with regard to performance of its contractors. On technical matters, NMC consulted with its subcontractors, but had the final authority related to KNPP decisions.

As licensee, NMC has responsibility for communication with the NRC and other regulatory agencies throughout the project regarding the related technical and licensing actions and soliciting assistance as necessary from members of the KNPP Uprate Project Team. NMC reviewed results of the analyses, evaluations and design and implementation of plant modifications, and incorporated them into the KNPP design and licensing basis. In select cases, NMC provided supporting analysis based on best engineering methods and practices available for use at the time. NMC will apply for licenses and license amendments as required to operate the plant at the uprated condition.

1.3.2 Westinghouse Electric Company

Westinghouse scope involves all NSSS-related analyses and evaluations, including the NSSS performance parameters, NSSS design transients, NSSS systems and components, design basis accidents, NSSS/BOP interface, containment, and reactor core nuclear fuel. As the KNPP Original Equipment Manufacturer (OEM) NSSS Designer/Supplier, Westinghouse has extensive historical design documentation and t engineering experience applicable to KNPP. Westinghouse has worked closely with NMC in the recent past on the RSG and Fuel Transition Programs. Therefore, many of the engineers assigned to the uprate project are familiar with KNPP facilities and have worked closely with NMC plant personnel. Also, as part of the above programs, NMC has performed rigorous technical and quality assurance audits of Westinghouse analyses, evaluations, and supporting engineering calculations in order to demonstrate compliance with NMC policy and procedures.

Westinghouse also had a partial scope balance-of-plant (BOP) effort as a sub-contractor to Stone & Webster.

1.3.3 Stone & Webster Power Corporation

Stone & Webster analyses and evaluations include the BOP systems and components, including radiological and environmental evaluations. Stone & Webster also reviewed the impact on station procedures.

The BOP scope of work includes engineering and associated review, evaluations, calculations, and analyses required to support the Thermal Power Uprate Project at the uprated NSSS power level of 1780 MWt. This work identifies impact and changes required to plant documentation and hardware, and demonstrate that the plant can operate safety, reliably, and meet regulatory requirements.

NSSS/BOP interface data were developed and exchanged between NMC, Westinghouse, SWPC, and Stone & Webster. This information formed the foundation for the BOP reviews, evaluations, calculations, and analyses associated with the following:

- BOP Systems and components
- Pipe stress and supports
- Structures
- Electrical
- Instrumentation and controls
- BOP radiological review
- Environmental assessment
- Generic issues and programs
- Plant procedures

1.3.4 Siemens-Westinghouse Power Corporation

The scope of effort performed by Siemens-Westinghouse Power Corporation (SWPC) includes an engineering study to evaluate the turbine generator for an uprating from 1650 MWt to 1780 MWt NSSS power. Two reports were prepared and present the results of the uprating study of the existing turbine configuration that will enable KNPP to pass the steam flow required for the targeted thermal cycle and to ensure that components meet SWPC mechanical and electrical design criteria at the new load.

1.3.5 Advanced Measurement and Analysis Group, Inc

The scope of effort provided by Advanced Measurement and Analysis Group, Inc (AMAG) includes the hardware and software required to support installation of the CROSSFLOW ultrasonic feedwater flow measurement system. The ultrasonic flow measurement (UFM) system will improve the measurement accuracy of the feedwater flow rate at KNPP and hence relax the requirement for a 2 percent power measurement uncertainty required by 10CFR Part 50, Appendix K. This is the basis for the measurement uncertainty recapture power uprate of 1.4 percent.

1.4 Methodology and Acceptance Criteria

1.4.1 Nuclear Steam Supply Systems

The methodology utilized in evaluation of the impact on the NSSS has been structured consistent with the methodology established in Westinghouse WCAP-10263, *A Review Plan for Uprating the Licensed Power of a PWR Power Plant*, dated 1983. Since submittal of WCAP-10263 to the NRC, the methodology has been used successfully as a basis for Power Uprate Projects on over 33 plants for a total of 1619 MWe of installed capacity. The Uprate Projects have ranged from a 1.0 percent to a 26.3 percent increase above base licensed power level.

The methodology in WCAP-10263 established the basis 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 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/parameters values, use of currently approved analytical techniques (for example, methodologies and computer codes) and use of currently applicable licensing criteria and standards.

The power uprate analyses and evaluations were performed in accordance with Westinghouse quality assurance requirements defined in the Westinghouse Quality Management System procedures, which comply with 10CR50 Appendix B criteria. These analyses and evaluations are in conformance with Westinghouse and industry codes, standards and regulatory requirements applicable to KNPP. Assumptions and acceptance criteria are provided in the appropriate sections of this report.

1.4.2 Balance of Plant

The methodology used for the evaluation of the BOP was the same as used successfully in many other Power Uprate Projects. The BOP systems, structures, and components were evaluated based on the existing design and licensing basis documented in the KNPP *Updated Safety Analysis Report* (USAR) and Technical Specification bases. Where the existing basis could not be met or a revised basis was used to demonstrate compliance to new criteria, justification for compliance and/or the revised basis is provided in the acceptance criteria used for the power uprate evaluation. In addition, evaluations were performed in areas where the existing documentation did not demonstrate capability at the uprated conditions, and summary results are provided in Section 9 of this report.

1.4.3 CROSSFLOW Ultrasonic Flow Measurement

The CROSSFLOW UFM is a non-intrusive device designed to be installed around existing feedwater piping without the need to make cuts or penetrations of any kind. The CROSSFLOW meter is a highly accurate device warranted to measure feedwater flow with an accuracy of 0.5 percent or better. This increased accuracy of feedwater flow measurement results in a reduced power measurement uncertainty. The design and operation of the system has been reviewed and approved by the NRC as detailed in *Safety Evaluation Report* (SER) issued on March 20, 2000, titled "Acceptance for Referencing of CENPD-397-P-A, Revision-01-P, Improved Flow Measurement Accuracy Using CROSSFLOW Ultrasonic Flow Measurement Technology."

1.4.4 Use of Computer Codes

The KNPP 7.4-percent power uprate and calorimetric measurement uncertainty recapture analyses and evaluations were performed using currently approved analytical techniques to demonstrate compliance with the licensing criteria and standards that apply to KNPP. In performing these analyses, methodologies and principal computer codes were used that are currently approved by the NRC for KNPP. The NRC-approved techniques are the same as those used for current KNPP analyses and are described in the KNPP USAR.

Table 1-1 contains a list of the principal computer codes used in each section of this Licensing Report. This table also delineates which codes have been previously used and approved for KNPP. Brief descriptions of the computer codes are provided in Table 1-2.

1.5 Technical Basis for Significant Hazards Evaluation

This report provides the technical basis for the significant hazards evaluation included with the associated Stretch Power Uprate License Amendment Request for KNPP.

1.6 Updated Safety Analysis Report Revisions

The KNPP *Updated Safety Analysis Report* (USAR) has been reviewed for necessary revisions initiated by the power uprate effort. At issuance of this licensing report, the associated USAR revision package was being developed and will be formally issued for use at the KNPP. These revisions should be incorporated into the KNPP USAR hardcopy with the USAR update that follows the NRC approval of the proposed stretch power uprate license amendment request.

1.7 Proprietary Information Designations

Westinghouse

There is information contained in this report that Westinghouse considers Westinghouse Proprietary. The specific information is contained within the brackets with designated superscripted letter (a through f), for example:

[Westinghouse Proprietary Information] ^{a,c}

The reason for marking Westinghouse Proprietary information in this report is so that if any portion of this report is used to prepare documents to be submitted to the NRC (for example, a licensing report), the authors will be aware of exactly which information is proprietary to Westinghouse and can protect the information accordingly. When a licensing report or any other document is submitted to the NRC for review, either the information proprietary to Westinghouse Electric Company LLC must be omitted from the submittal, or a non-proprietary version suitable for public disclosure must also be submitted.

1.8 Plant Impacts Due to Power Uprate

- No hardware modifications are required to the NSSS systems or components.
- Only one minor hardware modification, a change to valve trim in the FW control valve, is required to the BOP system components as a result of the power uprate.

- For the turbine generator, the following three minor modifications are recommended by the turbine manufacturer:
  - High-pressure outer cylinder horizontal joint bolting modification to accommodate the higher loading conditions.
  - The low-pressure-to-Jackshaft and low-pressure-to-generator coupling bolts should be replaced with bolts of higher strength material.
  - Overspeed trip settings will need to be reset to a slightly lower value to compensate for the overspeed response.

1.9 Conclusions

This KNPP Power Uprate Project document is a summary of how all plant NSSS and BOP systems and components, transient and accident analyses, containment and reactor core, and nuclear fuel, have been addressed to support a 7.4-percent uprated power condition at KNPP. The results of the NSSS and BOP analyses and evaluations satisfy the project purpose to demonstrate compliance with all applicable licensing criteria and requirements. Further, the evaluations and analyses have resulted in identification of plant modifications and operational impact. While minor in scope, the plant impacts have been adequately discussed and understood by all affected parties, and properly documented in accordance with plant policy and procedures. This document, in combination with referenced supporting documentation, form the basis for uprating KNPP by 7.4 percent.

Table 1-1 KNPP Power Uprate Project Westinghouse Computer Codes Used			
Report Section	**Analysis**	**Computer Code (1, 2)**	**Previously Used by KNPP**
4.3	Control Systems Operability – Margin to Trip Analysis	LOFTRAN (LOFT12)	Yes
5.2	Reactor Internals	WECAN	Yes
5.3	Fuel Assemblies	NKMODE WEGAP WECAN	Yes Yes Yes
5.5	RCS Piping and Supports	WESTDYN WESAN	Yes Yes
6.1	LB BELOCA	WCOBRA/TRAC COCO	Yes Yes
	SB LOCA	NOTRUMP/ SBLOCTA	Yes Yes
6.2	Non-LOCA Transients	ANC FACTRAN LOFTRAN PHOENIX-P RETRAN TWINKLE VIPRE	Yes Yes Yes Yes Yes Yes Yes
6.4	LOCA M&E	SATAN VI WREFLOOD EPITOME	Yes Yes Yes
6.4	Containment Integrity	GOTHIC 7.0p2	Pending (3)
6.5	MSLB M&E	LOFTRAN	Yes

Notes:

1. See Table 1-2 for a brief description of each code.
2. All codes listed are maintained under Westinghouse Configuration Control.
3. NMC Letter NRC-02-082 from Thomas Coutu to Document Control Desk, *Kewaunee Nuclear Power Plant Request for Use of GOTHIC 7 in Containment Design Basis Accident Analyses*, dated September 30, 2002 (TAC No. MB6408).

Table 1-1 (Cont'd)
KNPP Power Uprate Project
Westinghouse Computer Codes Used

Report Section	Analysis	Computer Code (1, 2)	Previously Used by KNPP
6.6	LOCA Hydraulic Forces	MULTIFLEX 3.0 LATFORC FORCE 2 THRUST	Yes Yes Yes Yes
7.1	Core Thermal-Hydraulic Design	THINC IV VIPRE	Yes Yes
7.2	Core Design	ANC PHOENIX-P	Yes Yes
7.3	Fuel Rod Design and Performance	PAD 3.4; PAD 4.0	Yes
7.4	Reactor Internals Heat Generation Rates	DORT/BUGLE-96	Yes
7.5	Neutron Fluence	DORT/BUGLE-96	Yes
7.6	Radiation Source Terms	ORIGEN2.1	Yes

Notes:

1. See Table 1-2 for a brief description of each code.
2. All codes listed are maintained under Westinghouse Configuration Control.

Table 1-2
Computer Code Description

ANC

ANC is an advanced nodal code capable of two-dimensional and three-dimensional neutronics calculations. ANC is the reference model for certain safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. In addition, three-dimensional ANC validates one-dimensional and two-dimensional results and provides information about radial (x-y) peaking factors as a function of axial position. It can calculate discrete pin powers from nodal information as well.

COCO

Calculation of containment pressure and temperature is accomplished by use of the digital computer code COCO. COCO is a mathematical model of a generalized containment. The proper selection of various options in the code allows the creation of a specific model for particular containment design. The values used in the specific model for different aspects of the containment are derived from plant-specific input data. The COCO code has been used and found acceptable to calculate containment pressure transients for many dry containment plants. Transient phenomena within the Reactor Coolant System (RCS) affect containment conditions by means of convective mass and energy transport through the pipe break.

For analytical rigor and convenience, the containment air-steam-water mixture is separated into a water (pool) phase and a steam-air phase. Sufficient relationships to describe the transient are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate boundary conditions. As thermo-dynamic equations of state and conditions may vary during the transient, the equations have been derived for all possible cases of superheated or saturated steam and subcooled or saturated water. Switching between states is handled automatically by the code.

Table 1-2 (Cont'd)
Computer Code Description

DORT/BUGLE-96

The DORT discrete ordinates transport module of the DOORS 3.1 code package, in conjunction with the BUGLE-96 cross-section library, is used to determine the neutron flux and gamma-ray heating rate environment. This code and the associated cross-section library have been used by Westinghouse to calculate vessel fluences and reactor internals heating rates for other projects that have been submitted to, and approved by, the Nuclear Regulatory Commission (NRC). Furthermore, these calculational tools are specified in Regulatory Guide 1.190 for this type of work.

EPITOME (see also SATAN-VI and WREFLOOD)

The EPITOME code continues the post-reflood portion of the transient from the time at which the secondary side 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 mass and energy release tables, and mass and energy balance tables with data at critical times. EPITOME is essentially an automated hand calculation.

FACTRAN

FACTRAN calculates the transient temperature distribution in a cross-section of a metal-clad UO_2 fuel rod and the transient heat flux at the surface of the cladding, using as input the nuclear power and the time-dependent coolant parameters of pressure, flow, temperature, and density. The code uses a fuel model that simultaneously contains the following features:

- A sufficiently large number of radial space increments to handle fast transients, such as a rod ejection accident
- Material properties that are functions of temperature and a sophisticated fuel-to-cladding gap heat transfer calculation
- The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the fuel

Table 1-2 (Cont'd)
Computer Code Description

FORCE2 (See also MULTIFLEX, LATFORC, and THRUST)

The FORCE2 program calculates the hydraulic forces that the fluid exerts on the vessel internals in the vertical direction by utilizing 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 conservation of linear momentum (one-dimensional). Note that the computed vertical forces in the LOCA forces analyses do not include body forces on the vessel internals, such as deadweight or buoyancy. The deadweight and other factors are part of the dynamic system model to which the LOCA forces are provided as an external load. When the vertical forces on the reactor pressure vessel (RPV) internals are calculated, pressure differential forces, flow stagnation on, and unrecoverable orifice losses across, and friction losses on, the individual components are considered. These force types are then summed together, depending upon the significance of each, to yield the total vertical force acting on a given component.

GOTHIC

GOTHIC solves the integral form of the conservation equations for mass, momentum, and energy for multi-component, two-phase flow. The conservation equations are solved for three fields; continuous liquid, liquid drops, and the steam/gas phase. The three fields may be in thermal non-equilibrium within the same computational cell. This would allow the modeling of subcooled drops (for example, containment spray) falling through an atmosphere of saturated steam. The gas component of the steam/gas field can comprise up to eight different non-condensable gases with mass balances performed for each component. Relative velocities are calculated for each field, as well as the effects of two-phase slip on pressure drop. Heat transfer between the phases, surfaces, and the fluid are also allowed.

The GOTHIC code is capable of performing calculations in three modes. The code can be used in the lumped parameter nodal network mode, the two-dimensional finite difference mode, and the three-dimensional finite difference mode. Each of these modes may be used within the same model.

Table 1-2 (Cont'd)
Computer Code Description

GOTHIC has been used to study hydrogen distributions, containment pressure and temperature transients, perform flow-field calculations for particle transport purposes, and surge-line flooding studies for loss of RHR cooling events during shutdown operations. The flexible noding and conservation equation solutions in the code allow its application to a wide variety of problems, not necessarily just containment pressure and temperature calculations.

LATFORC (See also MULTIFLEX, FORCE2, and THRUST)

The LATFORC computer code utilizes MULTIFLEX-generated field pressures, together with geometric vessel information (component radial and axial lengths), to determine the horizontal forces on the vessel wall and core barrel. The LATFORC code represents the vessel region with a model that is consistent with the model used in the MULTIFLEX blowdown calculation. 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 typically stored on magnetic tape and are calculated for the initial 500 msec of the blowdown transient. These forcing functions serve as required input in determining the resultant mechanical loads on primary equipment and loop supports, vessel internals, and fuel grids.

LOFTRAN

The LOFTRAN computer program is used for studies of transient response of a pressurized water reactor (PWR) system to specified perturbations in process parameters. LOFTRAN simulates up to four-loop systems by modeling the reactor vessel, hot- and cold-leg piping, steam generators (tube and shell sides), and pressurizer. The pressurizer heaters' spray, relief, and safety valves are also considered in the program. Point model neutron kinetics and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary sides of the steam generators utilize a homogeneous, saturated mixture for the thermal transients, and a water level correlation for indication and control. The Reactor Protection System simulation includes reactor trips on neutron flux, over-power and over-temperature, reactor coolant ΔT, high and low pressure, low flow, and high pressurizer level. Control systems, including rod control, steam dump, feedwater control, and pressurizer pressure controls are also simulated. The Safety Injection System (SIS), including the accumulators, is also modeled.

Table 1-2 (Cont'd)
Computer Code Description

LOFTRAN is a versatile program suited to accident evaluation and control studies as well as parameter sizing. It is also used in performing loss of normal feedwater anticipated transient without scram (ATWS) and loss-of-load ATWS evaluations.

LOFT12 is a single-loop version of LOFTRAN used for symmetric transients. LOFT12 was also used in the previous control systems analysis for KNPP.

Both single-loop and multi-loop codes have been approved by the NRC.

MULTIFLEX

The analysis for LOCA hydraulic forces used the NRC-approved MULTIFLEX computer code, which is the current Westinghouse analytical tool used for analyzing LOCA hydraulic forces. The code was used to generate the transient hydraulic forcing functions on the vessel and internals. This code was previously used for LOCA hydraulic forces analyses.

MULTIFLEX 3.0 is an engineering design tool that is used to analyze the coupled fluid-structural interactions in a PWR system during the transient following a postulated pipe rupture in the main RCS. The thermal-hydraulic portion of the MULTIFLEX code is based on the one-dimensional homogeneous model expressed in a set of mass, momentum, and energy conservation equations. These equations are quasi-linear, first-order, partial differential equations solved by the method of characteristics.

The employed numerical method utilizes an explicit time scheme along the respective characteristics. MULTIFLEX considers the interaction of the fluid and structure simultaneously, whereby the mechanical equations of vibration are solved through the use of the modal analysis technique. MULTIFLEX 3.0 generates the input for the post-processing codes LATFORC, FORCE2, and THRUST.

Table 1-2 (Cont'd)
Computer Code Description

NKMODE

NKMODE is used to establish an equivalent finite element model that will preserve the dynamic properties of the fuel assembly. Parametric studies of the assembly vibrational frequencies and mode shapes are performed using NKMODE. NKMODE calculates a set of equivalent spring-mass elements representing an individual fuel assembly structural system.

NOTRUMP/SBLOCTA

The approved codes for Appendix K small-break LOCA (SBLOCA) analyses are NOTRUMP and SBLOCTA. The NOTRUMP computer code is a state-of-the-art, one-dimensional general network code consisting of a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flow limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. Additional features of the code are condensation heat transfer model applied in the steam generator region, loop seal model, core reflux model, flow regime mapping, etc.

The SBLOCTA computer code is used to model the fuel rod response to the SB LOCA transient. It models three rods in the hot assembly (hot, average, and adjacent), modeling simultaneous radial and axial conduction. Other modeling features include various skewed axial power shapes, assembly blockage model due to clad swell, and rupture and zirc/water reaction.

NOTRUMP is used to model the thermal-hydraulic behavior of the system and thereby obtain time- dependent values of various core region parameters, such as system pressure, temperature, fluid levels and flow rates, etc. These are provided as boundary conditions to SBLOCTA. SBLOCTA then uses these conditions and various hot channel inputs to calculate the rod heatup, and ultimately, the peak clad temperature (PCT) for a given transient. Additional variables calculated by SBLOCTA are cladding pressure, strain, and oxidation.

Table 1-2 (Cont'd)
Computer Code Description

ORIGEN2.1 (See also FIPCO-V and TRICAL)

Fission product inventories were modeled with ORIGEN2, Version 2.1. 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.

PAD 3.4/4.0

The NRC-approved PAD code, with NRC-approved models for in-reactor behavior, is used to calculate the fuel rod performance over its irradiation history. PAD is the principal design tool for evaluating fuel rod performance. PAD iteratively calculates the interrelated effects of temperature, pressure, clad elastic and plastic behavior, fission gas release, and fuel densification and swelling as a function of time and linear power. Fuel rod design and safety analyses are based on updated values (up to 100-percent helium gas release) for the integral fuel burnable absorber (IFBA) helium gas release model.

PAD is a best-estimate fuel rod performance model, and in most cases the design criterion evaluations are based on a best-estimate-plus-uncertainties approach. A statistical convolution of individual uncertainties due to design model uncertainties and fabrication dimensional tolerances is used. As-built dimensional uncertainties are measured for some critical inputs, for example, fuel pellet diameter, and when available, can be used in lieu of the fabrication uncertainties.

PHOENIX-P

PHOENIX-P is a two-dimensional, multi-group transport theory computer code. The nuclear cross-section library used by PHOENIX-P contains cross-section data based on a 70-energy-group structure derived from ENDF/B-VI files. PHOENIX-P performs a two-dimensional, 70-group nodal flux calculation that couples the individual subcell regions (pellet, cladding, and moderator) as well as surrounding rods via a collision probability technique. This 70-group solution is normalized by a coarse energy group flux solution derived from a discrete ordinates calculation. PHOENIX-P is capable of modeling all cell types needed for PWR core design applications.

RETRAN

RETRAN is used for studies of transient response of a pressurized water reactor (PWR) system to specified perturbations in process parameters. This code simulates a multi-loop system by a lumped parameter model containing the reactor vessel, hot- and cold-leg piping, reactor coolant pumps, steam generators (tube and shell sides), main steam lines, and the pressurizer. The pressurizer heaters, spray, relief valves, and safety valves may also be modeled. RETRAN includes a point neutron kinetics model and reactivity effects of the moderator, fuel boron, and control rods. The secondary side of the steam generator uses a detailed nodalization for the thermal transients. The RPS simulated in the code includes reactor trips on high neutron flux, OTDT and OPDT, low RCS flow, high and low pressurizer pressure, high pressurizer level, and lo-lo steam generator water level. Control systems are also simulated including rod control and pressurizer pressure control. Parts of the Safety Injection System (SIS), including the accumulators, may be modeled. RETRAN approximates the transient value of DNBR based on input from the core thermal safety limits.

SATAN-VI (See also WREFLOOD and EPITOME)

The SATAN 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 thermo-dynamic 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.

Table 1-2 (Cont'd)
Computer Code Description

THINC IV

The THINC-IV computer program is used to determine coolant density, mass velocity, enthalpy, vapor void, static pressure, and departure from nucleate boiling ratio (DNBR) distributions along parallel flow channels within a reactor core under expected steady-state operating conditions. This code has had extensive experimental verification (Reference 1) and is considered a best-estimate code. The THINC-IV analysis is based on a knowledge and understanding of the heat transfer and hydro-dynamic behavior of the coolant flow and the mechanical characteristics of the fuel elements. The THINC-IV analysis provides a realistic evaluation of the core performance.

THRUST

The THRUST program calculates the hydraulic forces that the fluid exerts on the reactor coolant loop (RCL). The THRUST code uses the MULTIFLEX LOCA pressure transient as input in the calculation of the loop forces. In the THRUST computer code, the loop piping is represented by a series of control volumes. The pressure forces are calculated by THRUST wherever there are changes in either loop area or direction. The LOCA loop forces are then transmitted to the appropriate structural analysis group where they are then combined with the other design basis loads (that is, seismic, thermal, and system-shaking loads) where they are used to qualify the RCLs under the design basis loads.

TWINKLE

TWINKLE is a multi-dimensional spatial neutron kinetics code. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region, fuel-cladding-coolant heat transfer model for calculating point-wise Doppler and moderator feedback effects. The code handles up to 2000 spatial points and performs steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters as input, the code accepts basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. The code provides various output, for example, channel-wise power, axial offset, enthalpy, volumetric surge, point-wise power, and fuel temperatures. It also predicts the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.

Table 1-2 (Cont'd)
Computer Code Description

VIPRE

VIPRE-01 (VIPRE) is a three-dimensional subchannel code that has been developed to account for hydraulic and nuclear effects on the enthalpy rise in the core and hot channels. The VIPRE code is based on a knowledge and understanding of the heat transfer and hydro-dynamic behavior of the coolant flow and the mechanical characteristics of the fuel elements. The use of the VIPRE analysis provides a realistic evaluation of the core performance and is used in the thermal-hydraulic analysis.

The VIPRE core model is used with the applicable DNB correlations to determine DNBR distributions along the hot channels of the reactor core under all expected operating conditions. The VIPRE code is described in detail in Reference 2, including discussion on code validation with experimental data. The VIPRE modeling method is described in Reference 1, including empirical models and correlations used. The effect of crud on the flow and enthalpy distribution in the core is not directly accounted for in the VIPRE evaluations. However, conservative treatment by the VIPRE modeling method has been demonstrated to bound this effect in DNBR calculations.

WCOBRA/TRAC

WCOBRA/TRAC is a thermal-hydraulic computer code that is used in the approved best-estimate large-break LOCA (LBLOCA) methodology for the calculation of fluid and thermal conditions in a PWR during a LBLOCA.

WCOBRA/TRAC uses a two-fluid, three-field representation of flow in the vessel component. The three fields are a vapor field, a continuous liquid field, and an entrained liquid drop field. Each field in the vessel uses a set of three-dimensional continuity, momentum, and energy equations with one exception: a common energy equation is used by both the continuous liquid and the entrained liquid drop fields.

Table 1-2 (Cont'd)
Computer Code Description

The one-dimensional components consist of all the major components in the primary system, such as pipes, pumps, valves, steam generators, and the pressurizer. The one-dimensional components are represented by a two-phase, five-equation, drift flux model. This formulation consists of two equations for the conservation of mass, two equations for the conservation of energy, and a single equation for the conservation of momentum. Closure for the field equations requires specification of the interphase relative velocities, interfacial heat and mass transfer, and other thermo-dynamic and constitutive relationships. This best-estimate computer code contains the following features:

- Ability to model transient three-dimensional flows in different geometries inside the vessel
- Ability to model thermal and mechanical non-equilibrium between phases
- Ability to mechanistically represent interfacial heat, mass, and momentum transfer in different flow regimes
- Ability to represent important reactor components such as fuel rods, steam generators, RCPs, etc.

WECAN

The WECAN computer code is a general-purpose finite element code with capabilities including structural and thermal-hydraulic static and dynamic analyses. It is a direct descendent of the mainframe-version of the WECAN code that has been used in the nuclear industry since the early 1970s. It has been used by Westinghouse for safety-related work for many years on essentially all Westinghouse-provided NSSS analyses, such as core structural design (analyses including static, dynamic, and thermal), primary piping, primary equipment supports, primary equipment components, and spent fuel rack design.

Table 1-2 (Cont'd)
Computer Code Description

The WECAN computer program can be used to solve a large variety of structural analysis problems. These problems can be one-, two-, or three-dimensional in nature. It is capable of static elastic and inelastic analysis, steady-state hydraulic analysis, standard and reduced modal analysis, harmonic response analysis, and transient dynamic analysis.

The WECAN program is based on the finite element method of analysis. The analyst must model, or idealize the structure in terms of discrete elements and apply loadings and boundary conditions to these elements. The stiffness (or conductivity) matrix for each element is assembled into a system of simultaneous linear equations for the entire structure. This set of equations is then solved by a variation of the Gaussian elimination method known as the wave front technique. This type of solution makes it possible to solve systems with a large number of degrees of freedom using a minimum amount of core storage. The maximum number of allowed degrees of freedom in the wave front depends on the amount of core available, which in turn depends on the type of analysis being performed.

WECAN is organized in such a way that additional structural elements can be added with a minimum of effort. Input formats are similar for all elements and all types of analysis. Input used in the static analysis of a structure can be used for a dynamic analysis with only minor modifications.

WEGAP

WEGAP calculates the dynamic structural response of a pressurized water reactor core. WEGAP represents the transient structural response of one row of fuel assemblies, including impact at the grid elevation. With the appropriate analysis parameters such as grid impact stiffness and damping, the number of fuel assemblies in a planar array and gap clearance established, the WEGAP reactor core model is used for analyzing transient loadings.

Table 1-2 (Cont'd)
Computer Code Description

WESAN

The WESAN computer program was developed by Westinghouse to accomplish RCL equipment support structure analyses and evaluation. The evaluation is completed for the normal, upset, emergency and faulted plant operating conditions. The faulted evaluation is completed on a time-history basis.

Input, which is by punched cards (except the time-history LOCA loads, which are read from tape), includes the following:

1. Appropriate problem label and dimension statements
2. Six components of force on the structure for each of thermal, weight, pressure operational basis earthquake, safe shutdown earthquake, or LOCA loads
3. Member cross-sectional dimensions, type code, and material
4. A 6 x 6 member influence coefficient array for each end of each member

Forces or displacement loads on the structure are combined, transformed to the structure-coordinate system, and multiplied by member influences coefficients. The resulting member forces are then used with member properties in equations of stress and interaction to determine the adequacy of each member in the structure. All input data are included in the printed output along with maximum forces and stresses in the members.

WESTDYN

WESTDYN, a computer program used for the structural analysis of piping systems, calculates displacement, internal forces, and stress distributions in three-dimensional piping models, while subjecting them to static and dynamic loads.

The static analysis includes pressure, deadweight, thermal expansion, distributed and point loads, anchor motion, and uniformly applied accelerations.

The dynamic analysis includes seismic or hydro-dynamic response spectra and time-history dynamic analysis. The time-history dynamic analysis includes options for non-linear supports, support gaps, and unidirectional single acting restraints.

Table 1-2 (Cont'd)
Computer Code Description

In addition, WESTDYN utilizes post-processors for the stress analysis of ASME 1, 2, 3, or ANSI B31.1 piping, and also for generating support load summary sheets and equipment, and component qualification input data.

WESTDYN automatically calculates stress indices for standard ANSI fittings by user selection of the ASME piping evaluation code and edition. Allowable piping stress limits, coefficients of thermal expansion, and moduli of elasticity for a wide range of materials are also automatically calculated with user-supplied design and operating data.

WREFLOOD (See also SATAN-IV and EPITOME)

The WREFLOOD code is used for computing the reflood transient. It 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 Emergency Core Cooling System (ECCS) refills the reactor vessel and cools the core.

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, RCP 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 downcomer water levels, fluid thermo-dynamic conditions (that is, pressure, enthalpy, density) throughout the primary system, and mass flow rates through the primary system.

2.0 NUCLEAR STEAM SUPPLY SYSTEM PARAMETERS

The Power Uprate Project included Nuclear Steam Supply System (NSSS) performance analyses to develop bounding NSSS Performance Capability Working Group (PCWG) parameters for use in the analyses and evaluations of the NSSS, including parameters for NSSS design transients and analyses of systems, components, accidents, and nuclear fuel.

2.1 Performance Capability Working Group Parameters

2.1.1 Introduction and Background

The NSSS PCWG parameters are the fundamental parameters that are used as input in all the NSSS analyses. They provide the Reactor Coolant System (RCS) and secondary system conditions (temperatures, pressures, flow) that are used as the basis for the design transients and for systems, components, accidents, as well as fuel analyses and evaluations.

The PCWG parameters are established using conservative assumptions in order to provide bounding conditions to be used in the NSSS analyses. For example, the RCS flow assumed in generating the primary and secondary side conditions is the thermal design flow (TDF), which is a conservatively low flow that accounts for flow measurement uncertainty and assumes the maximum steam generator tube plugging (SGTP) level. The PCWG parameters were determined such that Nuclear Management Company (NMC) would have operating flexibility; therefore, a range of conditions was therefore established for the vessel average temperature (T_{avg}) and the SGTP level. The T_{avg} range was specified between 556.3° and 573.0°F, while the SGTP level can vary from 0 to 10 percent. An uprated NSSS power level of 1780 MWt and a TDF value of 89,000 gpm/loop were also used to generate the PCWG parameters.

2.1.2 Input Parameters and Assumptions

The major input parameters and assumptions used in the calculation of the PCWG parameters established for the Uprate Project are summarized below:

- The power level for the uprating was set at 1780 MWt NSSS (1772-MWt core). This is approximately 7.4 percent higher than the current NSSS power rating of 1657.1 MWt (1650-MWt core).
- The TDF of 89,000 gpm/loop was maintained.
- The SGTP values assumed were 0 and 10 percent at both ends of the Tavg range.
- Design core bypass flow was assumed to be 7.0 percent, which accounts for thimble plug removal.

- A range of full-power, normal-operating Tavg from 556.3° to 573.0°F was selected for the analyses.
- The parameters are applicable to the Model 54F steam generators.
- Feedwater temperature (T_{feed}) was increased from 427.3° to 437.1°F to correspond with the uprated power.
- Fuel Type: 14x14 422V+ Fuel. Note that the determination of NSSS parameters conservatively assumes a full core of this fuel type is in place. It is not necessary or appropriate to consider any transition core effects.

Some of the assumptions made for calculating the uprate NSSS parameters were done so in order that the parameters would remain bounded by those from the recent *Reload Transition Safety Report* (RTSR) (Reference 1) and Replacement Steam Generator (RSG) Programs. For example, the T_{avg} range was narrowed slightly from the previous range of 554.1 to 575.3°F to 556.3 to 573.0°F. By making this change, the maximum T_{hot} and minimum T_{cold} values are bounded by those from the previous programs, thereby reducing the need for as much analysis. The only NSSS parameters in Table 2.1-1 that are not bounded by those from the RTSR/RSG Programs are the NSSS and reactor power, feedwater temperature, and steam flow.

2.1.3 Discussion of Parameter Cases

Table 2.1-1 provides the NSSS design parameter cases generated and used as the basis for the Uprate Project.

Cases 1 and 2 were based on the minimum T_{avg} of 556.3°F with 0- and 10-percent SGTP, respectively. Likewise, Cases 3 and 4 were developed for the maximum T_{avg} of 573.0°F with 0- and 10-percent SGTP, respectively. Case 3 yielded the maximum secondary side steam conditions for analysis purposes, while Case 2 yielded the minimum secondary side steam conditions.

2.1.4 Acceptance Criteria

The primary acceptance criteria for the determination of the PCWG parameters were that they would not be so overly conservative that they penalized the Uprate Project analysis results, and that they would provide NMC with adequate flexibility and margin in plant operation.

2.1.5 Results and Conclusions

The resulting PCWG parameters, shown in Table 2.1-1, were used by Westinghouse in all the analytical efforts. Westinghouse performed the analyses and evaluations based on the parameter sets that were most limiting, so that the analyses would support operation over the range of conditions specified.

2.1.6 References

1. KEW-RTSR-02-021, *Kewaunee Nuclear Power Plant, RTSR Program*, RTSR, R. H. Owoc, July 19, 2002, attaches Sisk, R. B., *Reload Transition Safety Report for the Kewaunee Nuclear Power Plant*, July 2002 (NRC Submittal via NMC Letter NRC-02-067, *License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications. Conforming Technical Specification Changes for Use of Westinghouse VANTAGE+ Fuel* (Docket 50-305), July 26, 2002.

Table 2.1-1
Design Power Capability Parameters
Kewaunee 7.4-Percent Uprate

BASIC COMPONENTS			
Reactor Vessel, ID, in.	132	Isolation Valves	No
Core		Number of Loops	2
Number of Assemblies	121	Steam Generator	
Rod Array	14x14 422V+	Model	54F[2]
Rod OD, in.	0.422	Shell Design Pressure, psia	1100
Number of Grids	2I/5Z	Reactor Coolant Pump	
Active Fuel Length, in.	143.25	Model/Weir	93A/No
Number of Control Rods, FL	29	Pump Motor, hp	6000
Internals Type	RGE	Frequency, Hz	60

	Uprate Program[2]			
THERMAL DESIGN PARAMETERS	**Case 1**	**Case 2**	**Case 3**	**Case 4**
NSSS Power, % of Current Licensed Power	107.4	107.4	107.4	107.4
MWt	1780	1780	1780	1780
10^6 BTU/hr	6074	6074	6074	6074
Reactor Power, MWt	1772	1772	1772	1772
10^6 BTU/hr	6046	6046	6046	6046
Thermal Design Flow, Loop gpm	89,000	89,000	89,000	89,000
Reactor 10^6 lb/hr	69.34	69.34	67.87	67.87
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	7.0[3]	7.0[3]	7.0[3]	7.0[3]
Reactor Coolant Temperature, °F				
Core Outlet	595.5	595.5	611.3	611.3
Vessel Outlet	590.8	590.8	606.8	606.8
Core Average	560.2	560.2	577.1	577.1
Vessel Average	556.3	556.3	573.0	573.0
Vessel/Core Inlet	521.9	521.9	539.2	539.2
Steam Generator Outlet	521.6	521.6	538.9	538.9
Steam Generator				
Steam Temperature, °F	495.9	492.1	514.0	510.4
Steam Pressure, psia	656[1]	634[1]	771[1,4]	747[1]
Steam Flow, 10^6 lb/hr total	7.74	7.73	7.76	7.76
Feed Temperature, °F	437.1	437.1	437.1	437.1
Moisture, % max.	0.25	0.25	0.25	0.25
Steam Generator Tube Plugging, %	0	10	0	10
Zero Load Temperature, °F	547	547	547	547
HYDRAULIC DESIGN PARAMETERS				
Mechanical Design Flow, gpm			102,800	
Minimum Measured Flow, gpm total			186,000	

Notes:
1. Value includes a 19 psi steam generator internal pressure drop.
2. Parameters reflect Model 54F replacement steam generator.
3. Value accounts for thimble plug deletion
4. If a high steam pressure is more limiting for analysis purposes, a greater steam pressure of 809 psia, steam temperature of 519.4°F and steam flow of $7.77x10^6$ lb/hr total should be assumed.

3.0 NUCLEAR STEAM SUPPLY SYSTEM DESIGN TRANSIENTS

This section discusses the generation of Nuclear Steam Supply System (NSSS) and auxiliary equipment design transients for the uprated power conditions. Current NSSS design transients were analyzed for their continued applicability at uprated power and the resulting transient curves were provided to all system and component designers for use in their specific analyses. Section 3.1 describes the evaluation performed for NSSS design transients. Auxiliary equipment design transients were also evaluated to determine whether they remain applicable for use in the uprating analysis of all the auxiliary equipment in the NSSS. The results of this evaluation for auxiliary equipment design transients are presented in Section 3.2. The component evaluations are presented in Section 5 of this report.

3.1 Nuclear Steam Supply System Design Transients

3.1.1 Introduction and Background

As part of the original design and analyses of the NSSS components for the Kewaunee plant, NSSS design transients (that is, temperature and pressure transients) were specified for use in the analyses of the cyclic behavior of the NSSS components. The incorporation of the Model 54F replacement steam generator (RSG) and the inclusion of a T_{avg} operating window (Reference 1) necessitated revisions to these design transients. To provide the necessary high degree of integrity for the NSSS components, the transient 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 plant operating conditions. The transients selected for component fatigue analyses were representative of operating conditions that would be considered to occur during plant operations, and were considered to be sufficiently severe or frequent to be of possible significance to component cyclic behavior. The transients were selected to be conservative representations of transients that, when used as a basis for component fatigue analysis, would provide confidence that the component was appropriate for its application 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.

3.1.2 Input Parameters and Assumptions

NSSS design transients are based primarily on the NSSS Performance Capability Working Group (PCWG) parameters developed for the Power Uprate Program as discussed in Section 2 of this report. The NSSS PCWG parameters that were used in the development of the NSSS design transients for the RSG Program were compared to the NSSS PCWG parameters for the Power Uprate Program. The differences between the values used in the Kewaunee RSG Program and the Uprating Program were sufficient to reassess the NSSS design transients developed for the RSG Program and to require, if necessary, that revised NSSS design transients be specified for the power uprate.

3.1.3 Description of Analyses and Evaluations

Table 3.1-1 includes a comparison of the operating conditions for the two sets of plant configurations. Note that the uprating does not result in a major change in operating conditions from those for the RSG Program. Some of the key items for the design transients are the operating condition changes from 100-percent power to no-load. For the limiting cases, the existing parameter changes would not be affected or are affected only minimally. As part of the uprating effort, the full-power T_{avg} range was "clipped" or "tightened" from the existing design values (see Table 3.1-1 for existing/RSG versus Uprate Program values for high and low T_{avg}). This was done to maintain the highest value of full-power T_{hot} no greater than the existing design value, and the minimum value of T_{cold} no lower than the existing design value. The following provides an assessment of key variables:

T_{hot}: The limiting case is for the high T_{avg} condition. This shows the highest temperature change from no-load to 100-percent power. The change is 606.8°F at full power, to 547°F at no-load (low T_{avg} case shows a smaller change). This change is the same for the uprating as for the RSG Program.

T_{cold}: The limiting case is for the low T_{avg} condition. This shows the highest temperature change from no-load to 100-percent power. The change is 521.6°F at full power, to 547°F at no-load (high T_{avg} case shows a smaller change). This change is the same for the uprating as for the RSG Program.

T_{steam}: The limiting case is for the low T_{avg} condition. This shows the highest temperature change from no-load to 100-percent power. The change is 492.1°F at full power, to 547°F at no-load for the uprating. This change is slightly greater than the 494.0°F at full power, to 547°F at no-load noted for the RSG Program. Since the plant will be restricted to a steam pressure no less than 644 psia (saturated pressure at 494°F), this change is the same for the uprating as for the RSG Program.

Feedwater temperature (T_{feed}): The change is 437.1°F at full power, to 32°F (conservative value) at no-load for the uprating. This change is slightly greater than the 427.3°F at full power, to 32°F (conservative value) at no-load noted for the RSG Program.

From this comparison of the PCWG parameters for the Kewaunee RSG Program and the proposed uprate parameters, the differences were largely in the following parameters: steam generator T_{steam} (saturation temperature corresponding to steam generator steam temperature), and T_{feed}.

For the T_{feed} transient response, a design transient reanalysis was required. Since a lower limit of 644 psia is placed on the minimum full-power steam pressure (same minimum steam pressure noted for the RSG Program), then the steam generator steam temperature transient response would be bracketed by the transient response during all design transients included in the RSG effort. Therefore, for the plant uprating only the T_{feed} transient responses in the various design transients required revisions.

These Kewaunee-specific design transients have been used in the NSSS component and fatigue analyses and evaluations presented in Section 5 of this document.

3.1.4 Acceptance Criteria

There are no specific acceptance criteria.

3.1.5 Results and Conclusions

There are no specific results or conclusions for this section.

3.1.6 References

1. PCWG-2626, *Kewaunee (WPS): Approval of Category IIIP (for Partial Scope Contract) PCWG Parameters to Support RSG Model 54F MDF Increase and RTSR Program*, March 14, 2001.

Table 3.1-1

Comparison of Design Transient Analysis Conditions for RSG Program versus 7.4-Percent Uprate Program

Parameter	RSG Program (Reference 1)		Uprate Program (Section 2)	
	High T_{avg}	Low T_{avg}	High T_{avg}	Low T_{avg}
NSSS Power, MWt	1657.1	1657.1	1780	1780
Loop Flow Rate, gpm	89,000	89,000	89,000	89,000
T_{hot}, °F	606.8	586.3	606.8	590.8
T_{avg}, °F	575.3	554.1	573.0	556.3
T_{cold}, °F (steam generator exit)	543.6	521.6	538.9	521.6
No-Load Temperature, °F	547	547	547	547
Steam Flow, $x10^6$lb/hr total	7.14	7.11	7.76	7.73
Steam Generator Steam Pressure, psia	791	644	747	634
T_{steam}, °F	517.0	494.0	510.4	492.1
T_{feed}, °F	427.3	427.3	437.1	437.1

Note:

Parameters are for limiting 10-% steam generator tube plugging condition.

3.2 Auxiliary Equipment Design Transients

3.2.1 Introduction and Background

The Kewaunee auxiliary equipment design specifications included transients that were used to design and analyze the Class 1 auxiliary nozzles connected to the Reactor Coolant System (RCS) and certain Nuclear Steam Supply System (NSSS) auxiliary systems piping, heat exchangers, pumps, and tanks. These transients are described by variations in pressure, fluid temperature, and flow, and represent umbrella cases for operational events postulated to occur during the plant lifetime. To a large extent, the transients are based on engineering judgement and experience and are considered to be of such magnitude and/or frequency as to be significant in the component design and fatigue evaluation processes. The transients are sufficiently conservative so that, when used as the 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 auxiliary equipment design transients was performed based on the range of NSSS design parameters developed to support an NSSS power level of 1780 MWt (see Section 2 of this document).

The approved range of NSSS design parameters for the power uprate was compared with the NSSS design parameters used to develop the current design bases transients (Reference 1). The current design bases transients for Kewaunee are contained in *The Westinghouse Systems Standard Design Criteria* document (Reference 2).

3.2.3 Description of Analyses and Evaluation

An evaluation of the current design transients was performed to determine which transients could be potentially impacted by the power uprate. The evaluation concluded that the only design transients that could be potentially impacted by the power uprate are those temperature transients impacted by full-load RCS design temperatures.

These temperature transients are defined by the differences between the temperature of the coolant in the RCS loops and the temperature of the coolant in the auxiliary systems connected to the RCS loops. The greater the temperature difference, the greater the impact these temperature transients have on auxiliary component design and fatigue evaluation processes. Since the operating coolant temperatures in the auxiliary systems are not impacted by power uprate, the temperature difference between the coolant in the auxiliary systems and the coolant in the RCS loops is only impacted by changes in the RCS operating temperatures.

The current design temperature transients are based on a full-load hot-leg temperature (T_{hot}) of 630°F and a full-load cold-leg temperature (T_{cold}) of 560°F (Reference 1). These full-load temperatures were assumed for equipment design to ensure that the temperature transients would be conservative for a wide range of NSSS design parameters.

3.2.4 Acceptance Criteria and Results

A comparison of the range of NSSS design temperatures for power uprate at full-load, that is T_{hot} (590.8 to 606.8°F) and T_{cold} (521.9 to 539.2°F), with the T_{hot} and T_{cold} values used to develop the current design transients, indicates that the power-uprate temperature ranges are lower. These lower full-load operating temperatures result in less severe transients, since the temperature differences between RCS loop temperatures and the lower operating temperatures in the auxiliary systems connected to the RCS are less. For example, the temperature transients imposed on the Chemical and Volume Control System (CVCS) letdown and charging nozzles associated with starting and stopping letdown and charging flow would be less severe, since the temperature differences are less. Therefore, the current body of auxiliary design transients is conservative for the proposed power uprate.

3.2.5 Conclusions

The only auxiliary equipment transients that can be potentially impacted by the power uprate are those temperature transients related to full-load NSSS design temperatures. A review of these temperature transients indicates that if these transients were based on the power uprate design parameters, they would be less severe. Therefore, the current auxiliary equipment design transients for Kewaunee remain bounding for the proposed 7.4-percent NSSS power uprate.

3.2.6 References

1. KEW-RTSR-02-021, *Kewaunee Nuclear Power Plant, RTSR Program*, RTSR, R. H. Owoc, July 19, 2002, attaches Sisk, R. B., *Reload Transition Safety Report for the Kewaunee Nuclear Power Plant*, July 2002 (NRC Submittal via NMC Letter NRC-02-067, *License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications. Conforming Technical Specification Changes for Use of Westinghouse VANTAGE+ Fuel* (Docket 50-305), July 26, 2002.

2. *Westinghouse Systems Standard Design Criteria* (SSDC) document 1.3, Rev. 1, April 2, 1971.

4.0 NUCLEAR STEAM SUPPLY SYSTEMS

This chapter describes the results of the efforts performed in the Nuclear Steam Supply System (NSSS) area to support the uprating. Evaluations and analyses were performed to confirm that the NSSS continue to perform their intended functions under the uprated conditions. The systems addressed in this chapter are as follows:

Fluid Systems:

- Reactor Coolant System (RCS)
- Chemical and Volume Control System (CVCS)
- Residual Heat Removal System (RHRS)
- Safety Injection System (SIS)/Containment Spray System (CSS)
- Sampling System (SS)
- Component Cooling Water System (CCWS)

NSSS/Balance-of-Plant (BOP) Interface Systems:

- Main Steam System (MSS)
- Condensate and Feedwater (FW) System
- Auxiliary Feedwater System (AFS)
- Steam Generator Blowdown System (SGBS)

NSSS Control Systems:

- Pressure Relief Component Sizing
- Control Systems Setpoints Analysis

Detailed results and conclusions are presented within each subsection of this chapter.

4.1 Nuclear Steam Supply System Fluid Systems

4.1.1 Introduction and Background

As part of the evaluation for the Kewaunee Nuclear Power Plant (KNPP) 7.4-percent Power Uprate Project, the following fluid systems were reviewed to confirm continued compliance with industry codes and standards, regulatory requirements, and applicable performance and design basis requirements:

- Reactor Coolant System (RCS)
- Chemical and Volume Control System (CVCS)
- Residual Heat Removal System (RHRS)
- Safety Injection System (SIS)/Containment Spray System (CSS)
- Sampling System (SS)
- Component Cooling Water System (CCWS)

The fluid systems evaluations described in this section were performed at the system level. Evaluations of the Nuclear Steam Supply System (NSSS) components are described in Sections 5.1 through 5.9 of this report.

4.1.2 Input Parameters and Assumptions

The review was performed based on the approved range of NSSS design parameters shown in Section 2 of this document, which were developed to support a NSSS power level of 1780 MWt.

The approved range of NSSS design parameters were compared with the non-uprated design parameters previously evaluated for systems and components. The comparison indicated differences that could impact the performance of the above fluid systems. For example, the 7.4-percent power uprate would result in a proportional increase in the residual heat load after reactor shutdown that must be removed by the RHRS and CCWS during plant cooldown.

4.1.3 Acceptance Criteria

The evaluations of the above fluid systems relative to compliance with industry codes and standards, regulatory requirements, and applicable performance and design basis requirements are delineated in subsection 4.1.4. The acceptance criteria are included in subsection 4.1.4, along with the system evaluations, results, and conclusions.

4.1.4 Description of Fluid Systems Evaluations and Results

4.1.4.1 Reactor Coolant System

The changes in NSSS design parameters that impact the Reactor Coolant System (RCS) design bases functions include the increase in core power and the allowable range for average RCS temperature (T_{avg}). Verification that the major RCS components can support these changes is addressed in Sections 5.1 through 5.9 of this report. The increase in core power and the allowable RCS T_{avg} range also impacts the duty placed on the RCS control and protection systems. Verification that the RCS control and protection systems can support the power uprate is addressed in Sections 4.3 and 6 of this report. This section of the report will discuss the RCS supporting fluid system designs. These system designs include the pressurizer surge line, safety valves inlet and discharge piping, pressurizer relief tank, power-operated relief valve (PORV) inlet and discharge piping, pressurizer spray sub-system, resistance temperature detector bypass loop piping, and RCS instrumentation setpoints (excluding instrument channels used by the control and protection systems).

4.1.4.1.1 Pressurizer Surge Line, Safety Valves Inlet/Outlet Piping, and Pressurizer Relief Tank

The pressurizer safety valves are required to have adequate capacity to ensure that the RCS pressure does not exceed 110 percent of system design pressure. This is the maximum pressure allowed by the *American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code* for the worst-case loss-of-heat-sink event, that is, the loss of external electrical load. Based on the range of NSSS design parameters for the power uprate, an analysis of the loss-of-external-electrical-load transient was performed (see Section 6.2). The results of this analysis confirmed that the installed capacity of the pressurizer safety valves (690,000 lbs/hr or 345,000 lbs/hr/valve) is adequate to preclude RCS overpressurization. Based on results of this analysis, it can be concluded that the supporting fluid systems design of the surge line, safety valve inlet piping, and safety valve discharge piping is also adequate, since the design of these piping systems is based on safety valve design capacity.

The pressurizer relief tank (PRT) design (including the tank level setpoints) is also based on the total safety valve capacity, and conservatively sized to condense and cool a discharge of pressurizer steam equal to 110 percent of the steam volume above the full-power pressurizer program level. Since the loss-of-external-electrical-load transient analysis determined that the actual discharge of steam from the pressurizer into the PRT (540 lbm) is less than the design bases discharge (2965 lbm), the design of the PRT and existing level setpoints remain conservative. The current PRT level setpoints ensure adequate coolant is maintained in the tank to condense and cool the design bases discharge and preclude tank pressure from exceeding 50 psig. Therefore, the tank setpoints are conservative for the power uprate.

4.1.4.1.2 Pressurizer Power-Operated Relief Valves Inlet and Outlet Piping

The PORVs are required to have adequate capacity to prevent pressurizer pressure from reaching the high-pressure reactor trip setpoint for an external load reduction of up to 50 percent of rated electrical load. Based on the range of NSSS design parameters for the power uprate, a margin-to-trip analysis was performed (see Section 4.3 of this report). The results of this analysis confirmed that the installed capacity of the PORVs (358,000 lbs/hr, or 179,000 lbs/hr per valve at 2350 psia) is adequate to preclude a high-pressurizer pressure-reactor trip. Based on these results, it can also be concluded that the supporting fluid systems design of the PORVs inlet and discharge piping is also adequate, since this system piping is designed based on the design capacity of the PORVs.

4.1.4.1.3 Pressurizer Spray Sub-System

The pressurizer spray sub-system (valves and piping) is required to have adequate capacity to maintain the pressurizer pressure below the actuating set pressure of the PORVs (2350 psia), assuming a 10-percent step-load decrease from full power. Based on the range of NSSS design parameters for the power uprate, a margin-to-trip analysis was performed (see Section 4.3). The analysis concluded that the original design capacity of the spray sub-system (400 gpm or 200 gpm/valve) remains adequate for the 7.4-percent power uprate.

The pressurizer spray sub-system (valves and piping) was originally designed to pass the design spray valve flow rate (400 gpm or 200 gpm/valve) with an available pressure drop equal to the pressure drop from the spray flow scoop on each cold leg to the pressurizer surge line connection on the hot leg. The available pressure drop to achieve design spray flow must be based on minimum RCS loop flow, that is, thermal design flow (89,000 gpm), and the range of RCS design temperatures for T_{cold}. The most significant parameter is thermal design flow and this parameter is not impacted by the power uprate. An overall hydraulic evaluation concluded spray sub-system performance would be equal to, or greater than design for the full range of parameters approved for the power uprate.

4.1.4.1.4 Resistance Temperature Detector Bypass Loop Piping

The RCS fast-response temperature detectors that provide temperature signals to the reactor protection system are mounted in manifolds located in small bypass loops around the steam generator and the reactor coolant pump (RCP) of each loop.

The design of the reactor protection system requires that the fluid transport time from the reactor coolant loops (RCLs) to the last resistance temperature detector (RTD) in the RTD manifolds is less than or equal to 0.5 second. This limits the bypass loop piping size and length, and bypass flows to particular values. The bypass loops are sized to pass sufficient flow rates to meet this fluid transport delay time based on the available pressure drop in the main coolant loops.

With a given bypass loop configuration, the flow through the hot-leg bypass loops is primarily a function of the pressure drop across the steam generators and the flow through the cold-leg bypass loops is primarily controlled by the operating head of the RCPs. The available pressure drop to achieve required cold and hot leg loop bypass flows must be based on RCS flow and the range of RCS design temperatures for T_{cold} and T_{hot}. The most significant parameter is RCS thermal design flow (89,000 gpm) and this parameter is not impacted by the power uprate. An overall hydraulic evaluation concluded that loop bypass flows will remain equal to or greater than the values needed to achieve the required fluid transport time for the full range of parameters approved for the power uprate.

4.1.4.1.5 Reactor Coolant System Setpoints (Excluding Channels Used by the Control and Protection Systems)

The pressurizer spray line low-temperature alarm (TIA-422, 423) setpoint could potentially be impacted by the power uprate NSSS design parameters. The purpose of the alarm is to provide a warning to the operator if the miniflow through the spray lines is decreased. A fixed miniflow is required in the spray lines during normal operation to avoid undesirable temperature transients to the pressurizer spray nozzle and portions of the spray piping. Since the cold-leg temperature lower limit (521.9°F) is not significantly impacted by power uprate, there is no need to change the alarm setpoint.

4.1.4.2 Chemical and Volume Control System

The changes in NSSS design parameters that could potentially impact the CVCS design bases functions include the increase in core power and the allowable range for RCS full-load design temperatures. The increase in core power and the allowable range for RCS full-load design temperatures may affect the CVCS design bases requirements related to the core re-load boron requirements. Additionally, the allowable range for RCS full-load design temperatures may impact the heat loads that the CVCS heat exchangers must transfer to the CCWS, and in the case of the regenerative heat exchanger, to the charging flow.

System Heat Loads

The CVCS is designed to maintain RCS water inventory, boron concentration, and water chemistry. Other RCS support functions include purification and seal injection flow to the RCPs. During normal operation, the CVCS services the RCS by a letdown-and-charging process. RCS flow is letdown to the CVCS and delivered back to the RCS via charging pumps. This process requires that these feed-and-bleed streams be cooled and re-heated via heat exchangers. Since the power uprate alters RCS operating temperatures, the following CVCS heat exchangers were evaluated to assess the impact on the duty of these heat exchangers.

Regenerative Heat Exchanger

The regenerative heat exchanger is designed to recover heat from the letdown flow by reheating the charging flow to minimize RCS heat losses. Heating the charging line fluid before it enters the RCS is also required to minimize thermal transients on the RCS charging line nozzle. The design bases heat load was based on a maximum RCS T_{cold} temperature of 544.5°F. Since the maximum full-load T_{cold} (539.2°F) for the power uprate is less than 544.5°F, the heat exchanger duty is less severe than the design duty.

Letdown Heat Exchanger

The letdown heat exchanger cools the letdown flow, leaving the regenerative heat exchanger at a temperature that is compatible with the purification demineralizer resins and the RCP seals. The required temperature of the letdown flow (127.0°F) at the exit of the letdown heat exchanger is automatically controlled by a temperature controller and temperature control valve that regulates the flow of component cooling water that passes through the heat exchanger. The design bases heat load was based on a maximum inlet temperature of 290°F. Since the maximum inlet temperature for the power uprate is less than 290°F, the duty on this heat exchanger is less severe than the design duty.

Excess Letdown Heat Exchanger

The excess letdown heat exchanger can be employed either when normal letdown is temporarily out of service to maintain the plant in operation, or it can be used to supplement maximum letdown during the final stages of plant heatup. The design heat load was based on a maximum RCS T_{cold} temperature of 552°F. Since the maximum full-load T_{cold} (539.2°F) for the power uprate is less than design, the duty on this heat exchanger is less severe than the design duty.

Seal Water Heat Exchanger

The seal water heat exchanger is designed to cool the fluid from two sources:

- RCP seal water returning to the volume control tank (VCT)
- Reactor coolant discharged from the excess letdown heat exchanger

The first source of heat is not affected by the NSSS design parameters for the power uprate. The second heat source is reduced because the maximum RCS temperature (full-load T_{cold} 539.2°F) entering the excess letdown heat exchanger is less than the original design temperature (552°F). Therefore, the duty on the seal water heat exchanger for the power uprate is less severe than the design duty.

In summary, the heat duty on the CVCS heat exchangers is acceptable based on the NSSS design parameters approved for the power uprate.

4.1.4.2.2 Support of Reactor Coolant System (Core) Boron Requirements

The CVCS is designed to support the RCS (core) boron requirements. As part of the *Reload Transition Safety Report* (RTSR) (Reference 2), the CVCS functional and performance requirements related to boric acid storage and delivery were evaluated for the power uprate. The evaluation concluded the CVCS boron-related capability is adequate for power uprate.

4.1.4.3 Residual Heat Removal System

A higher power level results in an increase in the amount of residual heat being generated in the core during normal cooldown, refueling operations and accident conditions. This will result in a higher heat load on the residual heat exchangers during the cooldown and also during the refueling outage. The increased heat loads will be transferred to the CCWS and ultimately to the Service Water System (SWS). Evaluation of the performance of the RHRS in conjunction with the CCWS and SWS with the increased heat loads is addressed in the following subsections.

4.1.4.3.1 Normal Plant Cooldown

The RHRS is designed to reduce the temperature of the RCS to 140°F within 20 hours after reactor shutdown when the service water temperature is 66°F. The residual heat load was based on a core power of 1650 MWt. The RHRS is also capable of maintaining the RCS temperature at, or below 140°F when the RCS is open for refueling or maintenance operations. The design basis cooldown period was based on earlier plant design studies that concluded that 20 hours was an optimum cooldown time relative to the size and cost of the residual heat removal (RHR) and component cooling water (CCW) heat exchangers.

The RHRS is designed to operate 4 hours after reactor shutdown when the RCS pressure and temperature are 400 psig and 350°F, respectively. The maximum heat load removed by the RHRS occurs at this time. This heat load is the sum of the residual heat load produced by the core, the heat load generated by the operation of one RCP, and the sensible heat that must be removed from the RCS. The initial phase of plant cool-down is accomplished by employing the steam generators and condenser steam dump.

Since the residual heat load during plant cooldown would increase by 7.4 percent based on the uprated power, a cooldown analysis was performed to assess the impact of this added heat load on normal cooldown time. Normal cooldown is accomplished with two RHR heat exchangers and two CCW heat exchangers in service. Several cooldown scenarios were analyzed to assess the impact of power uprate on cooldown time. First, the increase in cooldown time was analyzed assuming design CCW flow of 1.25 M lb/hr to each RHR heat exchanger, design service water flow of 1.26 M lb/hr to each CCW heat exchanger, and the original design SWS temperature of 66°F. Additional analysis was performed to determine the impact of power uprate on cooldown time with reduced CCW flow (0.747 M lb/hr) to each RHR heat exchanger, reduced service water flow (1.13 M lb/hr) to each CCW heat exchanger, and a maximum SWS temperature of 80°F. These reduced CCW heat exchanger and RHR heat exchanger flows reflect current operating flows and the increased SWS temperature is currently reflected in the plant safety analysis (Reference 1). The impact of service water temperatures less than the current maximum of 80°F on cooldown time was also accessed with the current reduced flows to the RHR and CCW heat exchangers, since typically service temperature is expected to be significantly less than maximum allowable.

The results and conclusions of this normal plant cooldown analysis are as follows:

- For a core power uprate of 7.4 percent (from 1650 MWt to 1772 MWt), the Condenser Steam Dump System capacity is more than adequate to establish RHRS cut-in temperature and pressure within the current design basis cooldown time of 4 hours after reactor shutdown.

- Assuming design bases RHR and CCW heat exchanger flows, and the original design SWS temperature of 66°F, a core power uprate from 1650 MWt to 1772 MWt would increase the time for the RHRS to cool the plant down from 350° to 140°F from 13.9 hours to 17.5 hours, respectively. Therefore total cooldown time, that is, RHRS cooldown time plus 4 hours for steam dump, would increase from 17.9 hours to 21.5 hours, which exceeds the original design bases objective of 20 hours. This is considered acceptable since 20 hours was an economic objective.

- At the current reduced RHR and CCW heat exchanger flows and the current maximum allowable SWS temperature of 80°F, a core power uprate from 1650 MWt to 1772 MWt would increase the RHRS cooldown time from 57.2 hours to 76.2 hours respectively. Therefore, total cooldown time, that is, RHRS cooldown time plus 4 hours for steam dump, would increase from 61.2 hours to 80.2 hours. However, at service water temperatures less than 80°F, coupled with current reduced RHR and CCW heat exchanger flows, total cooldown time would be less than 80.2 hours. For example, at uprated power and a service water temperature of 66°F (original design), coupled with current reduced RHR and CCW heat exchanger flows, total cooldown time would be 37.5 hours.

These longer cooldown times are considered to be acceptable, considering the design bases for the 20 hours was simply economic, that is, minimizing the time required for cooldown versus the size and cost of the RHRS and CCWS components (that is, heat exchangers, pumps, etc.). Therefore, the RHRS, in conjunction with the CCWS and SWS, is adequately sized for normal cooldown heat loads associated with the power uprate.

4.1.4.3.2 10CFR Part 50, Appendix R, Fire Protection Report

In accordance with the KNPP Appendix R Fire Protection Report, the RHRS must also be capable of achieving RCS cold shutdown (200°F) in less than 72 hours after reactor shutdown, assuming the following:

- Loss-of-offsite power
- Single train of RHRS equipment, that is, one RHR pump and one RHR heat exchanger
- Single train of CCWS equipment, that is, one CCW heat exchanger and one CCW pump
- CCW flow (0.747 M lb/hr) to RHR heat exchanger, and service water flow (1.13 M lb/hr) to CCW heat exchanger reflect current operating flows
- Maximum SWS temperature (80°F) permitted by current safety analysis of record
- Initiation of RHRS operation at 29 hours after reactor shutdown, when a single train of RHRS and CCWS equipment can match core residual heat

The results of this Appendix R cooldown analysis shows that a single train of RHRS and CCWS equipment at uprated power can reduce the RCS temperature from 350° to 200°F in 40.2 hours, assuming the RHRS is placed in service 29 hours after reactor shutdown. Therefore, total cooldown time is 69.2 hours. This cooldown time is in compliance with the Appendix R cooldown time limit of 72 hours after reactor shutdown. Therefore, the RHRS, in conjunction with the CCWS and SWS, is adequately sized to meet the Appendix R regulatory requirements.

4.1.4.4 Safety Injection System/Containment Spray System

The required volume, duration, and heat rejection capability of the SIS and CSS flows in the event of a break are determined based on analytical and empirical models that simulate reactor and containment conditions subsequent to the postulated RCS and Main Steam System breaks. As a result of these analyses, the system and component criteria necessary to demonstrate compliance with regulatory requirements at the uprated power level are established. Since the results of these analyses (Section 6) have demonstrated that SIS/CSS provides adequate safety margin, the as-built SIS and CSS are acceptable for the power uprate.

4.1.4.5 Sampling System

The change in NSSS design parameters that potentially impact the Sampling System (SS) design bases is the allowable range for average RCS design temperature (T_{avg}). The change in RCS loop operating temperatures may affect the SS design bases requirement related to the maximum heat load that the SS heat exchangers must transfer to the CCWS.

4.1.4.5.1 System Heat Loads

The SS provides fluid samples from the RCS (pressurizer and hot leg) for laboratory analysis. The sample flows from the RCS are cooled (pressurizer steam samples condensed and cooled) via heat exchangers. Since the power uprate alters RCS loop operating temperatures, the SS heat exchangers were evaluated to assess the impact on the design duty of these heat exchangers.

4.1.4.5.2 Sampling System Heat Exchangers

There are three sample heat exchangers: one for the pressurizer steam sample, one for the pressurizer liquid sample, and one for the RCL hot leg. The design bases heat load, that is, the maximum heat load for sizing these heat exchangers, is based on condensing and cooling pressurizer saturated steam (652.7°F) down to 127°F. Since nominal pressurizer saturated steam temperature (652.7°F) is not impacted by the power uprate, the design duty assumed for the SS heat exchangers is not impacted by the power uprate.

4.1.4.6 Component Cooling Water System

The CCWS is an intermediate system between the RCS and the SWS. It ensures that leakage of radioactivity from the components being cooled is contained within the plant. Revised heat rejection rates and /or cooling water flow requirements were assessed due to the power uprate.

4.1.4.6.1 Heat Loads

The CCWS is designed to:

- Remove residual and sensible heat from the RCS via the RHRS during plant cooldown
- Cool the letdown flow to the CVCS during power operation
- Provide cooling to dissipate waste heat from various plant components
- Provide cooling to safeguards loads after an accident

The following primary and auxiliary equipment impose heat loads on the CCWS:

- Residual heat exchangers
- RCPs
- Letdown heat exchanger
- Excess letdown heat exchangers
- Seal water heat exchanger
- Boric acid evaporator
- Evaporator distillate cooler
- SS heat exchangers
- Waste gas compressors
- RHR pumps
- Safety injection pumps
- Containment spray pumps

The total CCWS heat load is variable depending on the plant operational mode and the equipment in service.

4.1.4.6.2 Plant Cooldown

The largest heat load on the CCWS occurs when the RHRS is placed in service during a normal plant cooldown. The increase in this transient heat load due to the higher residual heat load at the uprated power level was evaluated in conjunction with RHRS cooldown capability in subsection 4.1.4.3.1. The results of this evaluation concluded that the RHRS, in conjunction with the CCWS and SWS, is adequately sized for normal cooldown heat loads associated with the power uprate. Also, single-train cooldown based on 10CFR50, Appendix R fire protection requirements was evaluated (refer to subsection 4.1.4.3.2), and RHRS, in conjunction with the CCWS and SWS, was found to meet regulatory requirements.

4.1.4.6.3 Plant Heatup/Power Operation/Refueling

The heat loads imposed on the CCWS during plant heatup, power operation, and refueling are less limiting with respect to sizing the CCW heat exchangers. As noted previously in the CVCS and the SS evaluations, the majority of the heat loads imposed on the CCWS by these systems will either remain the same, or decrease. Also, during refueling, the heat load imposed on the CCWS by the residual heat exchanger(s) will increase due to the proportional increase in residual heat at the uprated power level. The RCP thermal barrier heat loads will decrease due to lower RCS cold-leg operating temperatures.

The changes in heat loads imposed on the CCWS during normal modes of plant operation are well within the system design capability.

4.1.4.6.4 Recirculation Phase of Safety Injection

In addition to normal cooldown, the RHR heat exchangers, in conjunction with the CCW heat exchangers, are also used to augment containment cooling during the SIS recirculation phase and terminate reactor coolant boiling in the longer term. During the SIS recirculation phase, the heat removal capability of the RHR heat exchangers is dependent upon CCWS supply temperature and sump water temperature, and will decrease with time as residual heat generation decreases. The maximum heat load imposed on the CCWS is based on maximum sump temperature and maximum CCWS supply temperature during the SIS recirculation phase. The containment integrity analysis defines the maximum recirculation heat load, and this heat load is acceptable from a CCWS design standpoint as long as the CCW supply temperature does not exceed the maximum recirculation limit of 130°F. The containment integrity analysis for power uprate confirms that the CCWS supply temperature will not exceed the maximum recirculation limit.

4.1.5 Conclusions

A summary of the conclusions of the evaluations of the NSSS fluid systems for the KNPP 7.4-percent Power Uprate Program is provided below.

4.1.5.1 Reactor Coolant System

The hydraulic design of the pressurizer surge line, safety valve inlet and outlet piping, and the design capacity of the PRT and existing level setpoints are adequate for the power uprate.

The hydraulic design of the pressurizer PORV inlet and discharge piping is adequate for the power uprate.

The hydraulic design of the pressurizer spray sub-system (piping and valves) is adequate to achieve design spray capacity at power uprate NSSS design parameters.

An overall hydraulic evaluation of the RTD bypass loop piping concluded that that loop bypass flows would be equal to, or greater than the values required to achieve the required fluid transport time for the full range of parameters approved for the power uprate.

The recommended spray line low-temperature alarm is not impacted based on the range of NSSS design parameters approved for power uprate.

4.1.5.2 Chemical and Volume Control System

The heat loads imposed on the CVCS heat exchangers due to the power uprate are less than the design basis heat loads.

As part of the RTSR (Reference 2), the CVCS functional and performance requirements related to boric acid storage and delivery were evaluated for power uprate. The evaluation concluded the following:

- CVCS boron-related capability is adequate for power uprate.

4.1.5.3 Residual Heat Removal System

The RHRS, in conjunction with the CCWS and SWS, is adequately sized for normal cooldown heat loads associated with the power uprate. The increase in plant cooldown time is considered acceptable based upon the economic benefits of the power uprate.

The Appendix R Fire Protection design basis cooldown requirements can be met at the uprated power level.

4.1.5.4 Safety Injection System/Containment Spray System

The as-built SIS and CSS are acceptable for the power uprate.

4.1.5.5 Sampling System

The heat loads imposed on the SS heat exchangers due to the power uprate are less than or equal to the design basis heat loads.

4.1.5.6 Component Cooling Water System

The CCWS, in conjunction with the RHRS and SWS, is adequately sized for normal cooldown heat loads associated with the power uprate. The increase in plant cooldown time is considered acceptable based upon the economic benefits of power uprate. Also, the CCWS in conjunction with the RHRS and SWS is adequately sized to meet the regulatory requirements of Appendix R with respect to cooldown.

The small changes in heat loads that are predicted to occur during normal modes of plant operation are well within the system design capability.

The containment integrity analysis defines the maximum recirculation heat load, and this heat load is acceptable from a CCWS design standpoint as long as the CCW supply temperature does not exceed the maximum recirculation limit of 130°F. The containment integrity analysis for power uprate confirms that the CCWS supply temperature will not exceed the maximum recirculation limit.

4.1.6 References

1. *Updated Safety Analysis Report, Kewaunee Nuclear Power Plant,* Rev. 17, FSAR Update, June 2002.

2. KEW-RTSR-02-021, *Kewaunee Nuclear Power Plant, RTSR Program,* RTSR, R. H. Owoc, July 19, 2002, attaches Sisk, R. B., *Reload Transition Safety Report for the Kewaunee Nuclear Power Plant,* July 2002 (NRC Submittal via NMC Letter NRC-02-067, *License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications. Conforming Technical Specification Changes for Use of Westinghouse VANTAGE+ Fuel* (Docket 50-305), July 26, 2002.

4.2 Nuclear Steam Supply System/Balance-of-Plant Interface Systems

4.2.1 Introduction and Background

As part of the KNPP 7.4-percent Power Uprate Project, the following balance-of-plant (BOP) fluid systems were reviewed to assess compliance with Westinghouse Nuclear Steam Supply System (NSSS)/BOP interface requirements (Reference 1):

- Main Steam System (MSS)
- Condensate and FW System
- Auxiliary Feedwater System (AFS)
- Steam Generator Blowdown System (SGBS)

The review was performed based on the range of NSSS design parameters shown in Section 2 of this document, which were developed to support a NSSS power level of 1780 MWt. The various interface systems were reviewed for the purpose of providing interface information that could be used in the more detailed BOP analyses. The results of those detailed analyses are provided in the BOP Engineering Report.

4.2.2 Input Parameters and Assumptions

The approved range of NSSS design parameters was compared with the non-uprated design parameters previously evaluated for systems and components. The comparison indicated significant differences that could impact the performance of the above BOP systems. For example, an increase in NSSS power of 7.4 percent (from 1657.1 MWt to 1780 MWt) and the approved lower limit on the average temperature (T_{avg}) (556.3°F) would result in approximately an 8.9-percent increase in steam/feedwater mass flow rates. Additionally, a steam generator tube plugging (SGTP) margin of 10 percent in combination with a T_{avg} in the upper end of the approved T_{avg} operating range (556.3° to 573.0°F) would result in a reduction of full-load steam pressure from 791 psia to 747 psia.

4.2.3 Acceptance Criteria

The NSSS/BOP system interface requirements are delineated in the *Westinghouse Steam Systems Design Manual* (Reference 1).

4.2.4 Description of Analyses and Evaluations

Evaluations of the above BOP systems relative to compliance with Westinghouse NSSS/BOP interface guidelines were performed to address the approved design parameters for the power uprate analyses that include ranges for parameters such as T_{avg} (556.3° to 573.0°F) and SGTP (0 to 10 percent average). These ranges of NSSS design parameters result in ranges on BOP parameters such as steam generator outlet steam pressure (634 psia to 809 psia). The NSSS/BOP interface evaluations were performed to address the impact of these ranges on NSSS and BOP parameters. The results of the NSSS/BOP interface evaluations are delineated below.

4.2.4.1 Main Steam System

The 7.4-percent power uprate, coupled with the potential reduction in full-load steam pressure to the design value of 747 psia, significantly impacts main steam-line pressure drop. At the design steam generator pressure of 747 psia, the full-load steam mass flow rate would increase about 8.7 percent; however, due to the reduced operating pressure and the lower-density steam, the volumetric flow rate would increase by approximately 15.5 percent, and steam-line pressure drop would increase by approximately 25.5 percent.

The NSSS design parameters for the current NSSS power (1657.1 MWt) resulted in a maximum steam-line pressure drop of about 35.4 psi and a minimum pressure of 608.6 psia at the turbine inlet valves. Based on the NSSS design parameters approved for the power uprating to an NSSS power of 1780 MWt, the lowest design steam generator pressure of 634 psia would result in a steam-line pressure drop of 42.5 psi, and a pressure at the turbine inlet valves of approximately 591.5 psia.

Operating the plant at the highest achievable full-load steam pressure can minimize the impact of main steam-line pressure drop on plant heat rate.

The following summarizes the Westinghouse evaluation of the major steam system components relative to the NSSS design parameters approved for the power uprate. The major components of the MSS are the steam generator steam safety valves, the steam generator power-operated atmospheric relief valves (ARVs), the main steam isolation valves (MSIVs) and non-return check valves, the condenser dump valves (CDVs), and atmospheric steam dump valves (ASDVs).

4.2.4.1.1 Steam Generator Safety Valves

The setpoints of the steam generator safety valves are determined based on the design pressure of the steam generators (1085 psig) and the requirements of the ASME Boiler and Pressure Vessel (B&PV) Code. Since the design pressure of the steam generators has not changed, there is no need to revise the setpoints of the safety valves.

The steam generator safety valves must have sufficient capacity to ensure that main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by the ASME B&PV code) for the worst-case loss-of-heat-sink event Section 6.2 of this report). Based on this requirement, Westinghouse applies the conservative criterion that the valves should be sized to relieve 100 percent of the maximum calculated steam flow at an accumulation pressure not exceeding 110 percent of the MSS design pressure (Reference 1). Additionally, the capacity of any single safety valve is presently limited to 890,000 lb/hr at 1100 psia based on the present steam line break analysis of record for a stuck-open steam generator safety valve (Reference 2).

The KNPP has ten safety valves with a total capacity of 7.660 x 10^6 lb/hr, which provide about 107.3 percent of the current maximum design full-load steam flow of 7.14 x 10^6 lb/hr. Based on the proposed range of NSSS design parameters approved for the power uprate, the installed safety valves provide about 98.6 percent of the maximum design steam flow of 7.77 x 10^6 lb/hr.

The plant safety analysis for the power uprate presented in Section 6.2, which summarizes non-LOCA event analyses documented in the *Reload Transition Safety Report* (RTSR) (Reference 3) confirms that the installed safety valve capacity of 7.660 x 10^6 lb/hr is adequate for overpressure protection.

4.2.4.1.2 Steam Generator Power-Operated Atmospheric Relief Valves

The steam generator ARVs, which are located upstream of the MSIVs and non-return check valves and adjacent to the steam generator safety valves, are automatically controlled by steam-line pressure during plant operations. The steam generator ARVs automatically modulate open and exhaust to atmosphere whenever the steam-line pressure exceeds a predetermined setpoint to minimize safety valve lifting during steam pressure transients. As the steam-line pressure decreases, the steam generator ARVs modulate closed and reseat at a pressure at least 10 psi below the opening pressure. The steam generator ARV set pressure for these operations is between zero-load steam pressure and the setpoint of the lowest-set steam generator safety valve. Since neither of these pressures change for the proposed range of NSSS design parameters, there is no need to change the ARV setpoint.

The steam generator ARVs also provide a means for decay heat removal and plant cool down by discharging steam to the atmosphere when either the condenser, the condenser circulating pumps, or steam dump to the condenser is not available. Under such circumstances, the ARVs, in conjunction with the AFS permit the plant to be cooled down from the pressure setpoint of the lowest-set steam generator safety valve to the point where the Residual Heat Removal System (RHRS) can be placed in service. During cool down, the ARVs are automatically controlled by steam-line pressure with remote manual adjustment of the pressure setpoint from the Control Room.

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

The steam generator ARVs are sized to have a capacity equal to about 10 percent of rated steam flow at no-load pressure (Reference 1). This capacity permits a plant cool down to RHRS operating conditions (350°F) in 4 hours (at an average rate of about 50°F/hr) assuming cooldown starts 2 hours after reactor shutdown. This sizing is compatible with normal cooldown capability and minimizes the water supply required by the AFS. This design basis is limiting with respect to sizing the ARVs and bounds the capacity required for tube rupture.

An evaluation of the installed capacity (745,000 lb/hr at 1050 psig) indicates that the original design basis in terms of cool down capability can still be achieved over the full range of NSSS design parameters approved for the power uprate.

4.2.4.1.3 Main Steam Isolation Valves and Non-Return Check Valves

The MSIVs in conjunction with non-return check valves are located outside the containment and downstream of the steam generator safety valves and ARVs. The valves prevent the uncontrolled blowdown of more than one steam generator and minimize the RCS cool down and containment pressure to within acceptable limits following a main steamline break (MSLB). To accomplish this, the MSIVs must be capable of closure within 5 seconds of receiving a closure signal against steam-line break-flow conditions in the forward direction.

Rapid closure of the MSIVs and non-return check valves following postulated steam line breaks causes a significant differential pressure across the valve seats and a thrust load on the MSS piping and piping supports in the area of the MSIVs and non-return check valves. The worst cases for pressure increase and thrust loads are controlled by the steam line break area (that is, mass flow rate and moisture content), throat area of the steam generator flow restrictors, valve seat bore, and no-load operating pressure. Since these variables and no-load pressure are not impacted by the power uprate, the design loads and associated stresses resulting from rapid closure of the MSIVs and non-return check valves will not change. Consequently, the power uprate has no significant impact on the interface requirements for the MSIVs or non-return check valves.

4.2.4.2 Steam Dump System

The NSSS RCS and the associated equipment (pumps, valves, heaters, control rods, etc.) were designed to provide satisfactory operation (automatic in the range of 15- to 100-percent power) without reactor trip when subjected to the following load transients:

- Loading at 5 percent of full power per minute with automatic reactor control.
- Unloading at 5 percent of full power per minute with automatic reactor control.
- Instantaneous load transients of plus or minus 10 percent of full power (not exceeding full power) with automatic reactor control.
- Load reductions of 100 percent of full power with automatic reactor control and steam dump.

The Steam Dump System creates an artificial steam load by dumping steam from ahead of the turbine valves to the condenser and the atmosphere. The Westinghouse original sizing criterion recommended that the steam dump system be capable of discharging 85 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 (Reference 1).

To prevent a trip, all NSSS control systems must be automatic. This includes the Rod Control System, which accommodates 10 percent of the load rejection.

For the power uprate, the maximum load reduction design basis is being relaxed from 100 percent to 50 percent of plant-rated power. The Westinghouse sizing criterion recommends that the Steam Dump System be capable of discharging 40 percent of the rated steam flow at full-load steam pressure to accommodate external load reductions up to 50 percent of plant-rated electrical load. Note that a steam-dump capacity of 40 percent of rated steam flow at full-load steam pressure also prevents steam generator safety valve lifting following reactor trip from full power.

4.2.4.2.1 Condenser/Atmospheric Dump Valves

The KNPP has six condensers and six ASDVs. The valves are sized to provide a total steam flow equal to 85 percent (40 percent to the condenser, and 45 percent to the atmosphere) of the original maximum calculated steam flow (7.5×10^6 lb/hr) at a full-load steam pressure of 750 psia. The valves are grouped into four banks, the first two banks discharging to the condenser, and the last two banks discharging to the atmosphere. At the current minimum full-load steam generator operating pressure of 644 psia, total steam dump is 5.48×10^6 lb/hr, or about 77 percent of rated steam flow (7.11×10^6 lb/hr). This reduced capacity relative to the original sizing criteria was verified to be adequate by a margin-to-trip analysis for load reductions up to 100 percent of rated electrical load. As noted above, for the power uprate, the maximum load reduction design basis is being relaxed from 100 percent to 50 percent of rated-electrical power. This relaxation should eliminate the need to credit the ASDVs to preclude reactor trip.

NSSS operation within the approved range of design parameters for power uprate at lower steam generator pressures and higher steam flows will result in a reduced steam dump capability relative to the current minimum capability. An evaluation indicates that the condenser steam dump capacity could be as low as 29.4 percent of rated steam flow (7.73×10^6 lb/hr), or 2.272×10^6 lb/hr, at a full-load steam pressure of 634 psia. These operating conditions are based on an assumed SGTP level of 10 percent and a T_{avg} in the lower end of the proposed operating range. If steam dump to the condenser is not adequate, credit can be taken for the actuation of the first bank of the ASDVs to supplement condenser dump, which would provide at least 40 percent steam dump. Therefore, total Steam Dump System capacity is adequate for load reductions up to 50 percent of electrical load.

The control systems margin-to-trip analysis (Section 4.3) concluded that condenser steam dump is adequate (that is, the ASDVs are not required) to preclude reactor trip for load reductions up to 50 percent of rated-electrical load.

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 not impacted by the power uprate.

4.2.4.3 Condensate and Feedwater System

The Condensate and FW System must automatically maintain steam generator water levels during steady state and transient operations. The range of NSSS design parameters for the power uprate will result in a required feedwater volumetric flow increase of up to 9.8 percent during full-power operation. The higher feedwater flow and temperatures will have an impact on system pressure drop, which may increase by as much as 19.5 percent. Also, a comparison of the range of NSSS design parameters for the power uprate with the non-uprated design parameters indicates that the steam generator full-power operating steam pressure may be increased by as much as 44 psi (771 psia to 815 psia).

The major components of the Condensate and FW System potentially impacted by the power uprate are the feedwater isolation valves (FIVs), the feedwater control valves (FRVs), feedwater bypass control valves (FBCVs), and the Condensate and FW System pumps.

4.2.4.3.1 Feedwater Isolation Valves

The FIVs are located outside containment and downstream of the FRVs. The valves function in conjunction with the primary isolation signals to the FRVs and backup trip signals to the feedwater pumps. This provides 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 FRVs and the backup FIVs must be capable of closure, following the receipt of any feedwater isolation signal.

The closure requirements imposed on the FRVs and the backup FIVs 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 line break from no-load conditions with the conservative assumption that both feedwater pumps are in service providing maximum flow following the break. Since these conservative assumptions are not impacted by the power uprate, the design loads and associated stresses resulting from rapid closure of these valves will not change.

4.2.4.3.2 Feedwater Control Valves, Feedwater Bypass Control Valves, Condensate, and Feedwater System Pumps

The Condensate and FW System available head in conjunction with the FRV 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. To ensure stable feedwater control with constant speed feedwater pumps, the pressure drop across the FRVs at rated flow (100-percent power) should be approximately equal to 1.5 times the FW System dynamic losses from the feed pump discharge through the steam generators. In addition, adequate margin should be available in the FRVs at full-load conditions to permit a Condensate and FW System delivery of 96 percent of rated flow with a 100-psi pressure increase above the full-load pressure with the FRVs fully open (Reference 1). This margin is required for load rejection. The FBCVs, provided for low-load operation of the FW System, are required to provide enough capacity to enable the plant to obtain 15-percent thermal power.

In light of these NSSS interface requirements, the Condensate and FW System piping, pumps, valves, and pressure-retaining components were evaluated to assess their ability to operate at the increased flow rates, temperatures, and pressures associated with the power uprate. The results are presented in the BOP Engineering Report.

To provide effective control of flow during normal operation, the FRVs and FBCVs are required to stroke open or closed within 20 seconds over the anticipated inlet pressure control range (approximately 0 to 1600 psig). This requirement is not impacted by the power uprate.

4.2.4.4 Auxiliary Feedwater System

The AFS supplies feedwater to the secondary side of the steam generators when the normal FW 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 cool-down operations, and also functions as an Engineered Safeguards System. In the latter function, the AFS is directly relied upon to prevent core damage and system over-pressurization 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 AFS are dictated by safety analyses, and the results of the revised safety analyses for the power uprate confirmed that the current AFS performance is acceptable.

4.2.4.4.1 Auxiliary Feedwater Storage Requirements

The AFS pumps are normally aligned to take suction from the condensate storage tanks (CSTs). To fulfill the engineered safety features (ESFs) 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 inventory requirements is the loss-of-all-AC-power transient. Since the Kewaunee plant is required to have a 4-hour coping period, the plant licensing basis requires that sufficient CST useable inventory must be available to bring the plant from full-power to hot-standby conditions and maintain the plant at hot standby for 4 hours. In light of this licensing basis, the plant current Technical Specifications require a minimum useable inventory of 39,000 gallons during power operation (Reference 4).

Since the minimum CST useable inventory is impacted by the power uprate (due to increased decay heat) and the approved range of NSSS operating conditions, a new analysis was performed as part of the BOP evaluations to determine the required inventory for the power uprate. This new analysis for the loss-of-all-AC-power scenario is based on the following conservative assumptions:

- Reactor trip occurs from 100.6 percent of rated core power (1772 MWt), from a low-low water level in the steam generators. A 2-second delay is assumed before reactor trip following loss of offsite power.
- Steam is released from the steam generators at the first safety valve setpoint plus setting tolerance and accumulation.
- The CST operating fluid temperature is at the maximum value (120°F for power uprate analyses).

The results of the analysis indicate that a minimum useable inventory should be increased from 39,000 gallons to 41,500 gallons to meet the loss-of-AC-power licensing basis for the range of NSSS operating conditions approved for power uprate.

4.2.4.5 Steam Generator Blowdown System

The SGBS controls the chemical composition of the steam generator secondary side water within the specified limits. The SGBS also controls the buildup of solids in the steam generator secondary.

The blowdown flow rates required during plant operation are based on chemistry control and tube-sheet sweep requirements to control the buildup of solids. The blowdown flow rate required to control chemistry and the buildup of solids in the steam generators is tied to allowable condenser in-leakage, total dissolved solids in the plant circulating water system, and allowable primary to secondary leakage. Since these variables are not impacted by the power uprate, the blowdown required to control secondary chemistry and steam generator solids will not be impacted by the power uprate.

The inlet pressure to the SGBS varies with steam generator operating pressure. Therefore, as steam generator full-load operating pressure decreases, the inlet pressure to the SGBS control valves decreases and the valves must open to maintain the required blowdown flow rate into the system. The present range of NSSS design parameters permits a maximum decrease in steam pressure from no-load to full-load of 376 psi (that is, from 1020 psia to 644 psia). Based on the revised range of NSSS design parameters approved for the power uprate, the no-load steam pressure (1020 psia) remains the same, and the minimum full-load steam pressure decreases 10 psi (to 634 psia). This small decrease is not considered to be significant with respect to blowdown flow control. Therefore, the range of design parameters approved for the power uprate will not impact blowdown flow control.

4.2.5 Conclusions

A summary of the conclusions of the evaluation of the NSSS/BOP system interfaces for the KNPP 7.4-percent NSSS Power Uprate Program is provided in the following subsections.

4.2.5.1 Main Steam System

Power uprating, coupled with reduced full-load steam pressures and corresponding higher steam-line pressure drops, will have a negative effect on the plant heat rate. It is recommended that the plant be operated at the highest achievable full-load steam pressure to minimize the plant heat rate.

The required safety valve capacity is dictated by safety analysis. The results of the safety analyses for the power uprate concluded that the installed safety valve capacity is adequate.

An evaluation of the capacity of the ARVs concluded that the original design basis in terms of cool-down capability can still be achieved over the full range of NSSS design parameters approved for the power uprate. This cooldown design basis, with respect to sizing the ARVs, is bounding in regard to the capacity required for tube rupture.

The NSSS/BOP interface systems requirements imposed on the design MSIVs, non-return check valves, and associated pipe loads are not impacted by the power uprate.

4.2.5.2 Steam Dump System

The KNPP Steam Dump System was originally sized to accommodate a steam flow equal to about 85 percent of the maximum calculated full-power steam flow (40-percent condenser dump and 45-percent atmospheric dump) to permit external load reductions up to 100 percent of the rated-electrical load. For the power uprate, the maximum required load reduction is being relaxed from 100 percent to 50 percent of the plant-rated power. This reduced load reduction capability results in a corresponding reduction in the maximum required steam-dump capacity from 85 percent to 40 percent of the rated-steam flow. In light of this change, the steam dump capacity for the range of NSSS design parameters approved for the power uprate exceeds the minimum recommended capacity of 40 percent of rated-steam flow for load reductions up to 50 percent of electrical load and remains acceptable for uprated conditions.

4.2.5.3 Condensate and Feedwater System

To ensure stable feedwater control, with constant speed feedwater pumps, the pressure drop across the FRVs at rated flow (100-percent power) should be approximately equal to 1.5 times the FW System dynamic losses from the feed-pump discharge through the steam generators. In addition, adequate margin should be available in the FRVs at full-load conditions to permit a Condensate and FW System delivery of 96 percent of the rated flow, with a 100-psi pressure increase above the full-load pressure with the FRVs fully open. This margin is required for load rejection. The FBCVs, provided for low-load operation of the FW System, are required to provide enough capacity to enable the plant to obtain 15-percent thermal power.

In light of these NSSS interface requirements, the Condensate and FW System piping, pumps, valves and pressure-retaining components were evaluated to ensure their ability to operate at the increased flow rates, temperatures, and pressures associated with the power uprate. The results are presented in Section 8.3.3.

The FRVs and FBCVs stroke time requirement (20 seconds) is not impacted by the power uprate.

4.2.5.4 Auxiliary Feedwater System

The minimum flow requirements of the AFS are dictated by safety analysis. The results of the revised safety analyses confirmed that the current AFS performance is acceptable for the power uprate.

The AFS pumps are normally aligned to take suction from the CSTs. In the event of a loss of all AC power, the plant licensing basis dictates that sufficient CST useable inventory must be available to bring the plant from full-power to hot-standby conditions and maintain the plant at hot standby for 4 hours. In light of this licensing basis, the plant Technical Specifications require a minimum useable inventory of 39,000 gallons during power operation. Since this minimum required inventory is impacted by the power uprate, a new analysis was performed for the power uprate. The results of the analysis indicates that a minimum useable inventory should be increased from 39,000 gallons to 41,500 gallons to meet the loss-of-AC-power licensing basis for the range of NSSS operating conditions approved for the power uprate.

4.2.5.5 Steam Generator Blowdown System

The blowdown flow rate required to control the chemistry and the buildup of solids in the steam generators is tied to allowable condenser in-leakage, total dissolved solids in the plant circulating water system, and allowable primary to secondary leakage. Since these variables are not impacted by the power uprate, the blowdown required to control secondary chemistry and steam generator solids will not be impacted by the power uprate.

The inlet pressure to the SGBS varies with steam generator operating pressure. Since the present range of the approved NSSS design parameters permits a range of operating pressures that almost bound the power uprate design parameters, the range of design parameters approved for the power uprate will not impact blowdown flow control.

4.2.6 References

1. WCAP-7451, *Westinghouse Steam Systems Design Manual*, February 1970.

2. *Kewaunee Nuclear Power Plant Updated Safety Analysis Report*, Rev. 17, June 1, 2002.

3. KEW-RTSR-02-021, *Kewaunee Nuclear Power Plant, RTSR Program*, RTSR, R. H. Owoc, July 19, 2002, attaches Sisk, R. B., *Reload Transition Safety Report for the Kewaunee Nuclear Power Plant*, July 2002 (NRC Submittal via NMC Letter NRC-02-067, *License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications. Conforming Technical Specification Changes for Use of Westinghouse VANTAGE+ Fuel* (Docket 50-305), July 26, 2002.

4. *Kewaunee Nuclear Power Plant Technical Specifications*, Revised through Amendment 160, May 28, 2002.

4.3 Nuclear Steam Supply System Control Systems

4.3.1 Pressure Relief Component Sizing

Based on the Nuclear Steam Supply System (NSSS) design parameters for the Kewaunee Power Uprating Program (Section 2), the installed capacities of the following NSSS pressure control components have been evaluated at the uprated conditions.

- Pressurizer PORVs
- Pressurizer spray valves
- Pressurizer heaters

The following concludes that the installed capacity of the listed pressure control components is still acceptable for the uprated conditions.

4.3.1.1 Pressurizer Power-Operated Relief Valves

4.3.1.1.1 Introduction and Background

The sizing basis for the pressurizer PORVs is to prevent the pressurizer pressure from reaching the high pressurizer pressure reactor trip setpoint for the design basis 50-percent load rejection with steam dump transient. For the Kewaunee plant, this is a turbine load rejection (that is, 200-percent/minute turbine runback) from 100- to 50-percent power.

4.3.1.1.2 Input Parameters and Assumptions

The pressurizer PORV sizing analysis was performed at the Kewaunee uprating design conditions defined in Section 2.1. The analysis was intended to bracket all operating conditions; full-power T_{avg} ranging from 556.3° to 573.0°F, and 0 to maximum steam generator tube plugging (SGTP) levels. The adequacy of the PORV relief capacity is most severely challenged for the operating condition that results in the largest surge flow into the pressurizer. The following assumptions were made in the analysis:

- Initial power level at 100.6 percent of full-uprated thermal power. Note that for conservatism, this accounts for 0.6-percent calorimetric error.

- Pressurizer PORV installed capacity: 179,000 lb/hr saturated steam per valve at 2335 psig, total of two valves.

- Pressurizer spray valves total capacity: 400 gpm.

- Pressurizer heater installed total capacity: 1000 kW, split between 792 kW in back-up heaters and 208 kW in proportional heaters.

- Control systems actuation logic and setpoints: NSSS control systems actuation logic and corresponding setpoints for rod control, steam dump control, pressurizer pressure and level control, and steam generator level control systems are included.

- Setting the protection system setpoint values or reactor trip delay time artificially high defeated protection system actuation.

- Best-estimate nuclear design parameters (moderator temperature coefficient, Doppler power defect, control rod worth, and startup data) are assumed. Conservatism was applied by assuming the moderator temperature coefficient was 0 at all times. In actuality, the full-power moderator temperature coefficient would be expected to be negative for all times in core life.

The key analysis assumptions for the pressurizer PORV sizing analysis are provided in subsection 4.3.1.1.3.

4.3.1.1.3 Description of Analysis and Evaluations

A 50-percent load rejection with steam dump transient was analyzed using a Kewaunee model of the configured version of the LOFTRAN computer code (Reference 1), which was used for the replacement steam generator (RSG) analysis. This Kewaunee model was used as a starting point for plant definition (Reactor Coolant System [RCS] volumes, power levels, RCS flow mixing coefficients, PORV/safety valve flows, pressurizer heater and spray capacities, etc.). Since the 50-percent load rejection transient is loop-symmetric, a single-loop version of the LOFTRAN code was used (this is the same LOFTRAN code version used in the original PORV sizing analysis). This computer code is a system-level program code and models the overall NSSS including the detailed modeling for control and protection systems. The method of analysis used was similar to the standard sizing procedure for pressurizer PORV original sizing calculations. The major analysis assumptions were as follows:

- A 50-percent load rejection with steam dump transient is initiated from 100.6 percent of full-uprated thermal power (100.6- to 50-percent load) in a step manner. Note that for conservatism, this accounts for 0.6-percent calorimetric error.

- The analysis was performed for the full-power high T_{avg} (573.0°F) condition with 0-percent SGTP as described in Section 2.1.

- A conservative initial steam generator mass 50 percent lower than nominal is assumed; this is a standard PORV-sizing analysis assumption.

- The initial pressurizer water level is nominal.

- Credit is taken for all normally active control systems, including pressurizer pressure and level control, steam dump control, and rod control systems.

4.3.1.1.4 Acceptance Criteria

The installed pressurizer PORVs capacity should limit pressurizer pressure to less than the fixed high pressurizer pressure reactor trip setpoint on a design basis 50-percent load rejection transient with subsequent automatic steam dump. This criterion is conservatively met if the total PORV capacity is greater than or equal to the peak pressurizer in-surge flow rate during this transient.

4.3.1.1.5 Results

For the analysis of a 50-percent load rejection at the high T_{avg} conditions, the maximum total relief capacity of 111,538 lb/hr of saturated steam at 2350 psia is required at the uprating conditions. The maximum total installed relief capacity of the pressurizer PORVs is 358,000 lb/hr at 2350 psia, which is sufficient to avoid actuating the high pressurizer pressure trip setpoint for the design basis 50-percent load rejection. The peak pressurizer pressure reached during this transient was 2351 psia, which is below the high pressurizer pressure reactor trip setpoint of 2400 psia. Therefore, the pressurizer PORV capacity is adequate for the uprating.

4.3.1.1.6 Conclusions

Since the installed capacity of 358,000 lb/hr saturated steam at 2350 psia is greater than the required relief capacity, the installed PORVs are adequate for the Kewaunee power uprating conditions.

4.3.1.2 Pressurizer Spray Valves

4.3.1.2.1 Introduction and Background

The sizing basis for the pressurizer spray valves is to prevent challenges to the pressurizer PORVs for a 10-percent step-load decrease transient. For load decreases up to 10-percent power, the spray valves are the sole means of controlling pressure without actuating the pressurizer PORVs when in automatic pressure control mode.

4.3.1.2.2 Input Parameters and Assumptions

The pressurizer spray valves sizing analysis was performed at the Kewaunee uprated design conditions defined in Section 2.1. Other key input parameters and assumptions are listed below.

- A 10-percent step-load decrease transient is initiated from 100.6-percent full-uprated thermal power. Note that for conservatism, this accounts for 0.6-percent calorimetric error. A transient initiated from full-power bounds all lower initial power levels.

- The pressurizer spray valve sizing analysis was performed for the limiting high T_{avg} (573°F), 0-percent plugging case. Similar to the PORV sizing analysis described in subsection 4.3.1.1, this analysis is an RCS heatup transient. Therefore, the same sensitivity results are expected as for the PORV sizing; the high T_{avg}, 0-percent plugging case would result in the largest pressurizer insurge and the greatest potential for challenging the pressurizer PORVs.

- A conservative initial steam generator mass of 90 percent of nominal is assumed; this is a standard PORV-spray-valve sizing analysis assumption.

- Steam dump control is disabled.

- Rod control, pressurizer spray control, pressurizer level control, and steam generator level control systems are assumed operational and in automatic mode. The pressurizer level control system is not explicitly modeled in the analysis since it is not critical to the analytical results. The transient pressurizer pressure response of concern (reaching the pressurizer PORV setpoint) happens too rapidly for pressurizer level control to impact the results.

- Best-estimate reactor kinetics (that is, moderator temperature coefficient, Doppler power defect, control rod worth, etc.) at BOL conditions are used. The moderator temperature coefficient was taken to be least negative for all possible core designs.

4.3.1.2.3 Description of Analysis and Evaluations

A 10-percent step-load decrease from full-power transient was analyzed using the Kewaunee model of the LOFTRAN code. The method of analysis is similar to the standard sizing procedure for pressurizer spray valves used in the original sizing calculations.

4.3.1.2.4 Acceptance Criteria

The design capacity (400-gpm total) for the pressurizer spray valves should limit pressurizer pressure to less than the pressurizer PORV actuation setpoint on a 10-percent step-load decrease transient. Per the Kewaunee uprate plant setpoints, this PORV actuation setpoint is 2350 psia.

4.3.1.2.5 Results

For the limiting case analyzed (high T_{avg}, 0-percent SGTP level), the results showed a maximum peak pressurizer pressure of 2301 psia at the limiting high T_{avg} of 573°F uprating design conditions.

4.3.1.2.6 Conclusions

Since the peak pressurizer pressure is less than the PORV actuation setpoint of 2350 psia, the installed capacity of 400 gpm is adequate.

4.3.1.3 Pressurizer Heaters

Pressurizer heater total capacity is proportional to pressurizer volume. The pressurizer heaters are sized on the basis of 1 kW/ ft^3 of pressurizer free volume. The required heater capacity is not affected by the uprating.

4.3.2 Control Systems Setpoints Analysis

4.3.2.1 Introduction and Background

Control systems operability analyses were performed on the NSSS control system setpoints for the Kewaunee plant, in order to determine that there is adequate margin to relevant reactor trip and engineered safeguard features (ESFs) actuation setpoints with the proposed 7.4-percent power uprate conditions. The conditions that were used as inputs to the analyses are provided in Section 2 of this report (PCWG parameters) and encompass a range of plant operating conditions. The following cases, at both high- and low-T_{avg} conditions, were analyzed:

- 50-percent load rejection from 100-percent power
- 10-percent step-load decrease from 100-percent power
- 10-percent step-load increase from 90-percent power

4.3.2.2 Input Parameters and Assumptions

The control systems operability analyses were performed at the Kewaunee uprating design conditions defined in Section 2.1. The analyses bracket all operating conditions; full-power T_{avg} ranging from 556.3° to 573.0°F, and 0- to 10-percent SGTP levels. The following assumptions were made for all normal transients analyzed:

- All applicable NSSS control systems are assumed to be operational and in the automatic mode of control (that is, rod control, steam dump control, pressurizer level, steam generator level control, and pressurizer pressure control).

- Best-estimate reactor kinetics parameters are modeled (that is, rod worth, moderator temperature coefficient, Doppler power defect, etc.). Since beginning-of-life (BOL) core physics parameters have lower differential rod worth and less negative moderator temperature coefficient, modeling BOL core characteristics typically yields more conservative results that would bound the full cycle of operation.

- Analysis of 10-percent tube plugging conditions bounds the 0-percent tube plugging conditions. Higher tube plugging is somewhat more conservative for short-term heatup transients due to a slower rate of heat transfer from the primary to secondary side of the plant. Furthermore, lower nominal steam temperatures and pressures reduce steam dump capacity during heatup transients, and reduce margin to steamline isolation (SI) actuation on low steam pressure during cooldown transients.

- The transient simulations are modeled to run for an 1800-second interval (30 minutes). Most challenges to the reactor trip and ESF actuation setpoints occur within the first minute of the design basis normal condition transients; it is anticipated that plant operators will respond to mitigate normal conditions transients in a much shorter time frame. Therefore, this simulation time frame is considered more than adequate for assessing control system response and stability considerations.

- The overtemperature and overpower ΔT reactor trip setpoints (OTΔT and OPΔT) that were used in these analyses are as follows:

OTΔT	OPΔT
$K_1 = 1.20$	$K_4 = 1.095$
$K_2 = 0.015$ /°F	$K_5 = 0.0275$ /°F
$K_3 = 0.00072$ /psig	$K_6 = 0.00103$ /psig

The time constants to the OTΔT and OPΔT equations are modeled as follows:

T_{avg} Lag	4.0 sec
ΔT lag	4.0 sec
OTΔT Lead/Lag on T_{avg}	30.0 / 4.0 sec
OPΔT Rate/Lag	10.0 sec
ΔT Lead/Lag	0.0 / 0.0 sec

- All other reactor trip and ESF actuation setpoints remain unchanged.

4.3.2.3 50-Percent Load Rejection from Full-Power Transient

4.3.2.3.1 Description of Analysis and Evaluations

A 50-percent load rejection with steam-dump transient was analyzed using the Kewaunee model of the LOFTRAN code (Reference 1). Since the 50-percent load rejection transient is loop-symmetric, a single-loop version of the LOFTRAN code was used. This computer code is a system-level program code and models the overall NSSS including the detailed modeling of the control and protection systems.

The 50-percent load rejection is the most severe operational transient that the plant would normally be subjected to without a reactor trip. A 100-percent load rejection was initially analyzed on a preliminary basis and the control systems could not support it without causing a reactor trip; therefore, the design basis 100-percent load rejection was revised to be a 50-percent load rejection. Thus, the transient is modeled as a turbine runback from 100.6- to 50-percent power at a maximum rate of 200 percent per minute. The RCS average temperature, secondary side steam temperature and pressure increase rapidly following this transient. The steam dump is available to the condenser, preventing both reactor trip and steam generator safety valve actuation. The resultant heatup causes a rapid increase in secondary side steam pressure and temperature, with a commensurate rise in reactor coolant average temperature and pressure. All NSSS control systems are available to mitigate this transient.

Only the first two banks of steam dump valves (condenser dump valves only) have been modeled in order to demonstrate that steam dump to the condenser is adequate to mitigate the transient without reactor trip. For a 50-percent load rejection, only these steam dump valves would be demanded to open.

4.3.2.3.2 Acceptance Criteria

The 50-percent load rejection from full power should provide adequate margins to the Technical Specifications reactor trip setpoints. The reactor trip setpoints that are important for this analysis were the high-power range nuclear flux setpoint, high pressurizer pressure setpoint, low pressurizer pressure setpoint, overtemperature ΔT (OTΔT) trip setpoint, overpower ΔT (OPΔT) trip setpoint, and the high pressurizer level setpoint. The plant response should be stable and non-oscillatory. There should be adequate pressurizer PORV capacity to prevent the transient from reaching the high pressurizer pressure reactor trip setpoint of 2400 psia (2385 psig).

4.3.2.3.3 Results

The results indicated that none of the above reactor trip setpoints are challenged. The control systems undergo one cycle of oscillations during the initial period of the 50-percent load rejection; however, the oscillation quickly dampens out as the transient progresses. This oscillation occurs due to the small temperature error (that is, T_{avg} minus T_{ref}) noted during the transient and the required high-steam dump load rejection controller gain needed to obtain acceptable results. However, no unstable parameters were noted and the plant response is acceptable.

The peak pressurizer pressure was controlled by the pressurizer PORV actuation, thereby preventing the pressurizer pressure from reaching the high pressurizer pressure reactor trip setpoint and showing acceptable capacity for the pressurizer PORVs.

The OTΔT and the OPΔT reactor trip setpoints, which are provided in subsection 4.3.2.2, were not reached for the load rejection transient; the minimum margin to the OTΔT and OPΔT trip setpoints were 4.9- and 6.4-percent, respectively. Therefore, the plant response is acceptable for the Power Uprating Program.

4.3.2.4 10-Percent Step-Load Decrease from Full-Power Transient

4.3.2.4.1 Description of Analysis and Evaluations

A 10-percent step-load decrease from full-power transient was analyzed using the Kewaunee model of the LOFTRAN code.

The 10-percent step-load decrease is initiated from 100.6-percent power. Note that for conservatism, this accounts for 0.6-percent calorimetric error. Secondary side steam pressure and temperature initially increase, lagged by primary average temperature and pressure increases. The power mismatch between the turbine load and nuclear power, and the resultant temperature error between T_{avg} and T_{ref} causes the rods to move into the core, reducing core power. Reactor coolant temperature and pressure are then restored to their equilibrium values.

This transient should not result in the pressurizer pressure reaching the pressurizer PORV actuation setpoint. Stability of the Rod Control System is also assessed.

4.3.2.4.2 Acceptance Criteria

During the 10-percent step-load decrease transient, the PORV actuation setpoint should not be challenged. Therefore, the maximum pressure reached during this transient should be below the PORV actuation setpoint of 2350 psia (2335 psig).

4.3.2.4.3 Results

The results indicated that no reactor trip setpoints are challenged and the control system response is stable and non-oscillatory. Pressurizer pressure reached a maximum of 2298 psia, and therefore the PORVs are not challenged. The plant response for this transient is acceptable.

4.3.2.5 10-Percent Step-Load Increase from 90-Percent Power Transient

4.3.2.5.1 Description of Analysis and Evaluations

A 10-percent step-load increase from 90-percent power transient was analyzed using the Kewaunee model of the LOFTRAN code.

The 10-percent step load increase is initiated from 90-percent power. Note, that for conservatism, a 0.6-percent calorimetric error was added to the final power level (that is, 100.6-percent). Secondary steam pressure and temperature decrease initially, followed by a primary T_{avg} and pressurizer pressure decrease. Pressurizer heaters are actuated to restore system pressure. The power mismatch between the turbine load and nuclear power, and the resultant temperature error between T_{avg} and T_{ref} causes the rods to move out of the core, increasing core power.

Since the 10-percent step-load increase transient will result in the lowest steam pressure of any of the operational transients, it is analyzed to demonstrate that the ESF actuation will not occur on low-steam pressure.

4.3.2.5.2 Acceptance Criteria

The 10-percent step increase is analyzed to demonstrate that ESFs actuation will not occur on low steam pressure. The low compensated steam pressure SI setpoint is 515 psia (500 psig) with a lead/lag of 12/2 seconds. Also for this transient, the high-power range nuclear flux setpoint, 109 percent of rated thermal power should not be challenged.

4.3.2.5.3 Results

The results indicated that no reactor trip or ESF actuation setpoints are challenged and the control system response is stable and non-oscillatory. Results also indicate that the minimum compensated steam pressure is 618 psia during the step-load increase initiated from the low T_{avg} operating condition. This minimum value provides adequate margin to the safety injection and steamline isolation setpoint of 515 psia (or 500 psig). Furthermore, the maximum high-power range nuclear flux that was reached was 101.9 percent of the rated thermal power; thus, there is adequate margin to the high-nuclear flux reactor trip setpoint.

4.3.2.6 Conclusions

The NSSS control systems setpoints for Kewaunee were reviewed for the uprate conditions and remain applicable for the uprate conditions. Results indicated adequate margin to protection systems setpoints and that the plant is adequately stable with the 7.4-percent power uprating operating conditions.

Furthermore, the following were concluded for all analyses:

- The pressurizer PORVs are not challenged for the 10-percent step-load decrease transient.
- There is adequate margin to the OPΔT and OTΔT reactor trip setpoints.
- There is adequate margin to high and low pressurizer pressure reactor trip setpoints for all normal condition transients analyzed.
- There is adequate margin to the ESF actuation setpoints for the 10-percent step-load increase from 90-percent power transient.

The control systems are stable and supportive of the 7.4-percent power uprating for all normal condition transients.

4.3.3 References

1. WCAP-7451, *LOFTRAN Code Description*, G. E. Heberle, Rev. 5, November 1989.

Table 4.3-1

Steam Dump Setpoints Used in Analysis

Parameter	High T_{avg}	Low T_{avg}
Deadband, °F	2.0	2.0
Full-Load T_{avg}, °F	573	556.3
No-Load T_{avg}, °F	547	547
Proportional Band, °F (full-load T_{avg} – no-load T_{avg} - deadband)	24.0	7.3
Controller Gain (1.0/proportional band)	0.0417	0.1370
Hi-1 Trip Open, °F	6.0	3.2
Hi-2 Trip Open, °F	14.0	5.7
Hi-3 Trip Open, °F	18.5	7.0
Hi-4 Trip Open, °F	26.0	9.3

5.0 NUCLEAR STEAM SUPPLY SYSTEM COMPONENTS

Evaluations were performed to determine the effects of the uprate parameters on the Nuclear Steam Supply System (NSSS) components. In general, the uprate-related input used for these evaluations are the PCWG parameters (Section 2) and the NSSS design transient changes (Section 3.1). Additional input parameters specific to particular components (for example, NSSS auxiliary equipment design transients for the auxiliary equipment evaluations) were considered and are discussed in the appropriate component evaluation section. The purpose of the evaluations performed for the NSSS components was to confirm that they continue to satisfy the applicable codes, standards, and regulatory guides under the uprate conditions.

Evaluations were performed in the following areas, and are described within the remainder of this section:

- Reactor vessel structural evaluations and integrity
- Reactor Pressure Vessel (RPV) System
- Fuel assemblies
- Control rod drive mechanisms (CRDMs)
- Reactor coolant loop (RCL) piping and supports
- Reactor coolant pumps (RCPs) and motors
- Steam generators
- Pressurizer
- NSSS auxiliary equipment

5.1 Reactor Vessel

5.1.1 Structural Evaluation

5.1.1.1 Introduction and Background

Evaluations were performed for the various regions of the Kewaunee reactor vessel to determine the stress and fatigue usage effects of Nuclear Steam Supply System (NSSS) operation at the revised operating conditions of the Kewaunee 7.4-percent Power Uprate Project throughout the current plant operating license. The evaluations concluded that the previous structural evaluations performed for the Kewaunee Replacement Steam Generator (RSG) Program continue to bound the effects of operation at the power uprate operating conditions.

5.1.1.2 Input Parameters and Assumptions

The evaluations assess the effects of the design transients on the most limiting locations with regard to ranges of stress intensity and fatigue usage factors. The evaluations consider a worst-case set of design transients from among the high-temperature power uprate conditions, the low-temperature power uprate conditions, and the original design basis. The Power Uprate Project reactor vessel normaloperating parameters (Section 2) are bounded by the reactor vessel operating parameters assumed for the RSG analysis. The design transients identified in Section 3 exhibited no changes that required assessment in addition to the RSG analysis. Furthermore, no new design interface loads were identified as a result of the Power Uprate Project.

5.1.1.3 Description of Analyses and Evaluations

Reactor vessel operation from plant startup through implementation of the power uprate and any future operation in accordance with the original design basis remain bounded by the stress and fatigue analyses. No additional revised maximum ranges of stress intensity and maximum usage factors were required for the Power Uprate Project. In all cases, the stress and fatigue evaluations performed for the previous RSG Program remain conservative so that no new calculational results were identified.

The evaluations for the RSG Program assessed the effects of the revised design transients and operating parameters on the most limiting locations with regard to ranges of primary-plus-secondary stress intensity and fatigue usage factors in each of the regions as identified in the reactor vessel stress report and addendum. The evaluations considered a worst-case set of operating parameters and design transients from among the high-temperature RSG conditions the lower temperature RSG conditions, and the original design basis. As a result of these considerations, all of the RSG parameter cases are fully covered by the evaluations, and reactor vessel operation in accordance with the Kewaunee RSG Project for the remainder of the current operating license was justified.

In addition, reactor vessel operation from plant startup until implementation of the RSG Project, and any future operation in accordance with the original design basis, is still fully covered by the stress and fatigue analyses in the reactor vessel stress report. Where appropriate, revised maximum ranges of stress intensity and maximum usage factors were calculated for the RSG Project. In other cases, the original design basis stress analysis remains conservative so that no calculations were necessary, and the maximum ranges of stress intensity and fatigue usage factors reported in the Combustion Engineering, Inc. (CE) stress report and addendum continue to govern.

In addition to the revised operating parameters and design transients for the RSG Program, a set of allowable loads was applied at the Kewaunee reactor vessel/reactor internals interfaces. The allowable loads were previously evaluated and justified for application to the Point Beach reactor vessels for their RSG Project. Based upon the close similarity between the Point Beach reactor vessels and the Kewaunee reactor vessel, the stresses resulting from the allowable loads were applied to the Kewaunee reactor vessel/reactor internals interfaces at the reactor vessel main closure flanges, outlet nozzles internal projections, and core support pads (lugs). The stresses due to the interface loads were combined by superposition with the thermal and pressure stresses due to normal operation from the Kewaunee reactor vessel stress report.

The evaluation of the reactor vessel for the RSG Project shows that it is acceptable for plant operation and these evaluations remain applicable in accordance with the 7.4-percent Power Uprate Project. Therefore, the reactor vessel power uprate evaluation addresses reactor operation with the operating temperature ranges and design transients discussed in Sections 2 and 3. Such operation is shown to be acceptable in accordance with the ASME Boiler and Pressure Vessel Code (Reference 1) for the remainder of the plant license.

5.1.1.4 Acceptance Criteria

The maximum range of primary-plus-secondary stress intensity resulting from normal and upset condition design transient mechanical and thermal loads should not exceed 3 S_m at operating temperature (Reference 1, Paragraph N-414.4).

The maximum cumulative usage factor resulting from the peak stress intensities due to the normal and upset condition design transient mechanical and thermal loads should not exceed 1.0 in accordance with the procedure outlined in Paragraph N-415 and N-416 of the ASME Boiler and Pressure Vessel Code (Reference 1).

5.1.1.5 Results

The reactor vessel power uprate evaluation demonstrates that the power uprate does not increase the ranges of stress intensity or cumulative fatigue usage factors for any of the various regions of the reactor vessel that were previously evaluated for the Kewaunee RSG Project. The maximum ranges of primary-plus-secondary stress intensity remain as evaluated and justified for the RSG Project. Additionally, the maximum cumulative fatigue usage factors reported for the original design basis are otherwise unaffected by the power uprate conditions and remain significantly below the acceptance criterion of 1.0.

Results from the RSG Project that remain applicable to the Power Uprate Project evaluation are discussed in the following paragraphs and are shown in Table 5.1-1.

The RSG Program affected several of the maximum ranges of primary-plus-secondary stress intensity reported in the Kewaunee reactor vessel stress report. The evaluations show that for the limiting locations, most of the maximum ranges are unchanged when the revised operating parameters, design transients, and design interface loads are incorporated. The exceptions are the outlet nozzles, the closure studs, the control rod drive mechanism (CRDM) housings, the inlet nozzles, and the bottom head instrumentation tubes. The maximum cumulative fatigue usage factors at all of the limiting locations except the CRDM housings, the vent nozzle, the vessel wall transition, and the bottom head instrumentation tubes increase somewhat from the values reported. Most of the usage factor increases are minimal, but some of them are relatively large. Most importantly, all of the cumulative fatigue usage factors remain under the 1.0-limit. The greatest increases in usage factor are [ ]^{a,c} in the vessel flange, [ ]^{a,c} in the safety injection nozzles, [ ]^{a,c} in the external support brackets, and [ ]^{a,c} in the closure studs. These large increases resulted from the application of higher upper-head temperatures and reactor vessel/reactor internals interface loads, as well as the RSG design transients. The maximum calculated cumulative usage factor in the reactor vessel is [ ]^{a,c} in the external support brackets. The external support bracket cumulative usage factor increased from [ ]^{a,c} to this maximum value of [ ]^{a,c}, due solely to the effect of the revised design transients.

The maximum range of stress intensity at the outlet nozzle safe end increased by [ ]^{a,c} ksi to [ ]^{a,c} ksi, which is less than the ASME Section III $3S_m$ of 49.2 ksi. The maximum range of stress intensity for the low alloy steel nozzle increased by [ ]^{a,c} ksi, but the revised value of [ ]^{a,c} ksi remains less than the $3S_m$ limit of 80.1 ksi. The maximum range of stress intensity at the inlet nozzle safe end increased by [ ]^{a,c} ksi to [ ]^{a,c} ksi. This result remains below the $3S_m$ limit of 52.9 ksi. The maximum range of stress intensity for the low-alloy steel portion of the inlet nozzle increased by [ ]^{a,c} ksi to [ ]^{a,c} ksi, and remains less than the limit of 80.1 ksi. The greatest change is in the reported maximum stud service stress, which increased by [ ]^{a,c} ksi to [ ]^{a,c} ksi, but remains less than the $3S_m$ allowable of 118.8 ksi. This increase in the maximum service stress was largely due to the consideration of a 10-percent contingency that was added to the stud preload.

Allowable reactor vessel/reactor internals interface loads were incorporated using loads and stresses that were previously calculated for Point Beach. The reported maximum ranges of primary-plus-secondary stress intensity for the CRDM housings and the bottom head instrumentation tubes are actually reduced. The reduction is due to the exclusion of the steam pipe break transient from consideration in the primary-plus-secondary stress evaluation. Steam pipe break is a faulted condition and need not be included in the range of stress intensity evaluation. Exclusion of the steam pipe break keeps the maximum ranges of stress intensity for the CRDM housings and instrumentation tubes below the $3S_m$ limit, and eliminates the need for the simplified elastic-plastic analyses that are in the CE stress report.

The updated maximum ranges of primary-plus-secondary stress intensity and maximum cumulative fatigue usage factors for the Kewaunee reactor vessel accounting for the RSG Program are the results provided in Table 5.1-1. These results are unchanged by the 7.4-percent power uprate.

The RSG evaluation for the reactor vessel is documented in an addendum to the Kewaunee reactor vessel stress report (Reference 2). Based upon the satisfactory results of the evaluations in this addendum report as previously discussed, the Kewaunee reactor vessel is acceptable for plant operation in accordance with the Kewaunee RSG Project.

The effects of the Kewaunee 7.4-percent Power Uprate Project normal operating parameters and design transients are bounded by the corresponding effects considered in the reactor vessel evaluation for the Kewaunee RSG Project. Thus, considering any combination of the design basis, the power uprate, and the RSG NSSS design transients for the specified numbers of occurrences, the Kewaunee reactor vessel stress and fatigue analyses and evaluations justify operation with a range of vessel outlet temperature from 586.3°F up to 606.8°F, and a range of vessel inlet temperature (T_{cold}) from 521.9°F up to 543.8°F. Therefore, the reactor vessel evaluation for the RSG Project, in conjunction with the reactor vessel stress report, bounds the effects of reactor operation in accordance with the 7.4-percent Uprate Project with the parameters in Section 2. Such operation was considered acceptable in accordance with the 1968 Edition of Section III of the ASME Boiler and Pressure Vessel Code with Addenda through the Winter 1968 (Reference 1) for the remainder of the plant license.

5.1.1.6 Conclusions

The reactor vessel structural evaluation concludes that all acceptance criteria continue to be met for the Power Uprate Project.

5.1.2 Reactor Vessel Integrity-Neutron Irradiation

5.1.2.1 Introduction and Background

Reactor vessel integrity is affected by any changes in plant parameters that affect neutron fluence levels or temperature/pressure transients. Note that the temperature/pressure transients have been reviewed and do not change as a result of the Kewaunee Nuclear Power Plant (KNPP) Power Uprate Program. The balance of subsection 5.1.2 addresses the potential impact of the neutron fluence increase, resulting from the KNPP Power Uprate Program, on reactor vessel integrity.

5.1.2.2 Input Parameters and Assumptions

Uprated Fluence Projections

The calculated fluence projections (Section 7.5 of this document) on the vessel were evaluated for the uprated power level for input into the reactor vessel integrity evaluations. Typically, fluence values are used to evaluate end-of-life (EOL) transition temperature shift (EOL ΔRT_{NDT}) for development of surveillance capsule withdrawal schedules, determining EOL USE values, adjusted reference temperature (ART) values for determining applicability of heatup and cooldown curves and low-pressure overpressure protection (LTOP) limitations, Emergency Response Guideline (ERG) limits, and RT_{PTS} values. The calculated fluence projections used in the Power Uprate Program evaluation comply with Regulatory Guide 1.190 (Reference 3). As these calculations are performed on a plant-by-plant basis, there is no generic topical for approved method—the methodology used is that of Regulatory Guide 1.190 (Reference 3).

Inlet Temperature

Due to the uprated power, the reactor vessel inlet temperature may also change. This updated inlet temperature (Section 2 of this document) has been reviewed to verify its compliance with Regulatory Guide 1.99, Revision 2 (Reference 4). The reactor vessel inlet temperature at uprated power is evaluated to verify its compliance with Regulatory Guide 1.99, Revision 2 (Reference 4).

Low-Temperature Overpressure Protection System

The Low-Temperature Overpressure Protection System (LTOPS) provides protection against violations of the reactor vessel Appendix G pressure versus temperature limits during hot- or cold-shutdown operation at relatively low Reactor Coolant System (RCS) temperatures (generally below about 350°F). This protection is provided via a spring-loaded relief valve with a pressure setpoint to avoid violation of these Appendix G limits for the design basis transients.

The existing LTOPS setpoint would have to be revised for one or more of the following reasons:

- A change in the Nuclear Steam Supply System (NSSS) resulting in a change in the RCS volume

- A change in the steam generators, resulting in a change in the heat transfer coefficient during shutdown conditions
- A change in the design basis mass input or heat input transients
- A change in the spring-loaded relief valve, flow capability, etc.
- A change in the reactor vessel Appendix G pressure versus temperature limits used as an upper pressure limit that the LTOPS must protect against

5.1.2.3 Description of Analyses and Evaluations

The reactor vessel integrity evaluation for the Kewaunee uprating included the following objectives. The acceptance criteria and results for each of these objectives are discussed in more detail under subsections 5.1.2.4 and 5.1.2.5.

- Review of the reactor vessel surveillance capsule removal schedule to determine if changes are required as a result of changes in the vessel fluence due to the Power Uprate Program.
- Review of the existing P-T limit curves to determine if a new applicability date needs to be calculated due to the effects of the uprated fluence projections.
- Review of the existing RT_{PTS} values to determine if the effects of the uprated fluence projections result in an increase in RT_{PTS} for the beltline materials in the Kewaunee reactor vessel at the EOL (33 EFPY) and EOLE (51 EFPY).
- Review the ERG limits to determine if the applicable ERG category would change.
- Review the USE values at EOL for all reactor vessel beltline materials in the Kewaunee reactor vessel to assess the impact of the uprated fluence projections.
- Review reactor vessel inlet temperature for Kewaunee to verify that it maintains an acceptable level after the uprated condition takes effect.
- Review the LTOP limitations to determine if a new applicability date needs to be calculated due to the effects of the uprated fluence projections.

Surveillance Capsule Withdrawal Schedules

A surveillance capsule withdrawal schedule is developed to periodically remove surveillance capsules from the reactor vessel. This procedure is followed to effectively monitor the condition of the reactor vessel materials under actual operating conditions. The current surveillance capsule schedule defined in the KNPP USAR is consistent with the recommended number of surveillance capsules and withdrawal schedule cited in ASTM E185-82 (Reference 7). It is noted that the KNPP surveillance capsule program is based on ASTM E185-70 (Reference 5) and WCAP-8107 (Reference 6), *Wisconsin Public Service Corp. Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program.* Testing is performed in accordance with the methods specified and additional provisions identified in ASTM E185-82 (Reference 7).

The surveillance capsule withdrawal schedule is in terms of EFPY of plant operation, with a design life of 33 EFPY. Other factors that must be considered in establishing the surveillance capsule withdrawal schedule are maximum fluence values at the vessel inside surface and 1/4-thickness (1/4T) location.

The first surveillance capsule is typically scheduled to be withdrawn early in vessel life to verify initial predictions of surveillance material response to the actual radiation environment. It is generally removed when the predicted shift exceeds expected scatter by sufficient margin to be measurable. Normally, the capsule with the highest lead factor is withdrawn first. Early withdrawal also permits verification of the adequacy and conservatism of reactor vessel P-T operational limits.

The withdrawal schedule for the maximum number of surveillance capsules to be withdrawn is adjusted by the lead factor so that:

- Exposure of the *second* surveillance capsule withdrawn occurs when the accumulated neutron fluence of the capsule corresponds to a value midway between that of the first and third capsules.
- Exposure of the *third* surveillance capsule withdrawn does not exceed the peak EOL 1/4T fluence.
- Exposure of the *fourth* surveillance capsule withdrawn does not exceed the peak EOL reactor vessel fluence.

- Exposure of the *fifth* surveillance capsule withdrawn does not exceed twice the peak EOL reactor vessel fluence.

Per ASTM E185-82 (Reference 7), the four steps used for development of a surveillance capsule withdrawal schedule are as follows:

- Estimate peak vessel inside surface fluence at EOL and the corresponding transition temperature shift (ΔRT_{NDT}). This identifies the number of capsules required.

- Obtain the lead factor for each surveillance capsule relative to peak beltline fluence.

- Calculate the EFPY for the capsule to reach peak vessel EOL fluence at the inside surface and 1/4T locations. These results are used to establish the withdrawal schedule for all but the first surveillance capsule.

- Schedule surveillance capsule withdrawals at the nearest vessel refueling date.

A current surveillance capsule withdrawal schedule for the Kewaunee reactor vessel is documented in the KNPP USAR and WCAP-14279 (Reference 8). This schedule has been evaluated for the Power Uprate Project due to increased neutron fluence. In addition, the supplemental requirements imposed under the Master Curve Process (Reference 11) were also reviewed and are not impacted by the KNPP Power Uprate Program.

Heatup and Cooldown Pressure-Temperature Limit Curves

A review of the applicability dates of the heatup and cooldown curves, for Kewaunee, was performed. The curves are currently contained in WCAP-14278, Revision 1 (Reference 9). This review was carried out by comparing the fluence projections used in the current calculations of adjusted reference temperature for all the beltline materials in the Kewaunee reactor vessels to the fluence based on the uprated condition in Section 7.5 of this document.

Pressurized Thermal Shock

Pressurized thermal shock (PTS), a limiting condition on reactor vessel integrity, is postulated to occur during a severe system transient such as a loss-of-coolant-accident (LOCA) or a steam line break. Such transients may challenge the integrity of a reactor vessel under the following conditions:

- Severe overcooling of the inside surface of the vessel wall followed by high repressurization
- Significant degradation of vessel material toughness caused by radiation embrittlement
- The presence of a critical-size defect in the vessel wall

The PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may produce the propagation of flaws postulated to exist near the inner wall surface, thereby, potentially affecting the integrity of the vessel.

In 1985, the Nuclear Regulatory Commission (NRC) issued a formal ruling on PTS. It established screening criterion on PWR vessel embrittlement as measured by nil-ductility reference temperature, termed RT_{PTS}. RT_{PTS} screening criteria values were set (using conservative fracture mechanics analysis techniques) for beltline axial welds, plates, and beltline circumferential weld seams for EOL plant operation. All PWR vessels in the U.S. have been required to evaluate vessel embrittlement in accordance with these criteria through EOL.

The NRC amended its regulations for light water nuclear power plants to change the procedure for calculating radiation embrittlement. The revised PTS Rule was published in the Federal Register, December 19, 1995, with an effective date of January 18, 1996 (Reference 10). This amendment makes the procedure for calculating RT_{PTS} values consistent with the methods given in Regulatory Guide 1.99, Revision 2 (Reference 4).

The Rule establishes the following requirements for all domestic, operating PWRs:

- For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee should have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material.
- The assessment of RT_{PTS} must use the calculation procedures given in the PTS Rule and must specify the bases for the projected value of RT_{PTS} for each beltline material.

The report must specify the copper and nickel contents and the fluence values used in the calculation for each beltline material.

- This assessment must be updated whenever there is a significant change in projected values of RT_{PTS} or upon the request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are significant if either the previous value or the current value, or both values, exceed the screening criteria prior to expiration of the operating license, including any renewal term (if applicable), for the plant.

The RT_{PTS} screening criteria values for the beltline region are:

270°F for plates, forgings and axial weld materials, and
300°F for circumferential weld materials.

RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. Equation 1 must be used to calculate values of RT_{NDT} for each weld, plate or forging in the reactor vessel beltline.

$$\text{Equation 1:} \quad RT_{NDT} = RT_{NDT(U)} + M + \Delta RT_{NDT}$$

Where,

$RT_{NDT(U)}$ = Reference temperature for a reactor vessel material in pre-service or non-irradiated condition

M = Margin added to account for uncertainties in values of $RT_{NDT(U)}$, copper and nickel content, fluence, and calculation procedures. M is evaluated from Equation 2.

$$\text{Equation 2:} \quad M = \sqrt{\sigma_U^2 + \sigma_\Delta^2}$$

σ_U is the standard deviation for $RT_{NDT(U)}$.

σ_U = 0°F when $RT_{NDT(U)}$ is a measured value.

σ_U = 17°F when $RT_{NDT(U)}$ is a generic value.

σ_Δ is the standard deviation for RT_{NDT}. σ_Δ is not to exceed one half of ΔRT_{NDT}

For plates and forgings:

σ_{Δ} = 17°F when surveillance capsule data is not used.

σ_{Δ} = 8.5°F when surveillance capsule data is used.

For welds:

σ_{Δ} = 28°F when surveillance capsule data is not used.

σ_{Δ} = 14°F when surveillance capsule data is used.

ΔRT_{NDT} is the mean value of the transition temperature shift, or change in RT_{NDT}, due to being irradiated and must be calculated using Equation 3.

$$\text{Equation 3:}\quad \Delta RT_{NDT} = (CF) * f^{(0.28-0.10\log f)}$$

CF (°F) is the chemistry factor, which is a function of copper and nickel content. CF is determined from Tables 1 and 2 of the PTS Rule (Reference 8). Surveillance data deemed credible must be used to determine a material-specific value of CF. A material-specific value of CF is determined in Equation 5.

f is the calculated neutron fluence, in units of 10^{19} n/cm^2 (E > 1.0 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence. The 33 EFPY EOL fluence are used in calculating RT_{PTS}.

Equation 4 must be used for determining RT_{PTS} using Equation 3 with EOL fluence values for determining RT_{PTS}

$$\text{Equation 4:}\quad RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS}$$

To verify that RT_{NDT} for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to reactor vessel operating temperature and any related surveillance program results. Results from the plant-specific surveillance program must be integrated into the RT_{NDT} estimate if the plant-specific surveillance data has been deemed credible. According to Regulatory Guide 1.99, Revision 2 (Reference 4), to use surveillance data there has to be "...two or more credible data sets...from the reactor in question." A material-specific value of CF for surveillance materials is determined from Equation 5.

$$\text{Equation 5: } CF = \frac{\sum [A_i * f_i^{(0.28-0.10\log f_i)}]}{\sum [f_i^{(0.56-0.20\log f_i)}]}$$

In Equation 5, "A_i" is the measured value of ΔRT_{NDT} and "f_i" is the fluence for each surveillance data point. If there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld, that is, differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of RT_{NDT} must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance weld.

It should be noted here that the above methodology was used for evaluating the KNPP Power Uprate Program for all the beltline materials except the intermedfiate to lower shell circumferential weld. In this case, the Master Curve technology was used as described in WCAP-15075 (Reference 11), and with modifications as described in the NRC's Safety Evaluation (Reference 12) of the request for exemption by Kewaunee. Equation 6 was used to determine RT_{PTS} for the intermediate to lower shell circumferential weld using Master Curve technology.

$$\text{Equation 6: } RT_{PTS} = RT_{TO} + M + Bias$$

Where,	RT_{TO}	=	T_O + 33°F
	M	=	Margin = 62.5°F
	Bias	=	8.5°F

(All the above terms were prescribed or adjusted by the NRC in Reference 12.)

New T_o values and adjusted reference temperatures have been calculated for the intermediate to lower shell circumferential weld using the methodology described in Reference 12 corresponding to EOL and EOLE fluence for the KNPP power uprate.

Emergency Response Guideline Limits

Emergency Response Guideline (ERG) P-T limits were developed to establish guidance for operator action in the event of an emergency, such as a PTS event. Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT} at EOL. These generic categories were conservatively generated for the Westinghouse Owners Group (WOG) to be applicable to all Westinghouse plants.

The highest EOL RT_{NDT} for which the generic category ERG pressure-temperature limits were developed is 250°F for a longitudinal flaw and 300°F for a circumferential flaw. Thus, if the limiting vessel material has an EOL RT_{NDT} that exceeds 250°F for a longitudinal flaw or 300°F for a circumferential flaw, plant-specific ERG pressure-temperature limits are required.

A comparison of the current RT_{PTS} calculation (which is the EOL RT_{NDT} value [33 EFPY]) has been made to the uprated RT_{PTS} values for Kewaunee to determine if the applicable ERG category would change. The ERG categories are presented in Table 5.1-5.

Upper Shelf Energy

The integrity of the reactor vessel may be affected by changes in system temperatures and pressures resulting from the power uprate. To address this consideration, an evaluation was performed to assess the impact of the power uprate on the USE values for all reactor vessel beltline materials in the Kewaunee reactor vessel. The USE assessment used the results of the neutron fluence evaluation for the power uprate and Figure 2 of Regulatory Guide 1.99, Revision 2 (Reference 4), to determine if a further decrease in USE at EOL would occur due to the effects of the uprate on fluence projections.

Inlet Temperature

Paragraph 1.3.2 of Regulatory Guide 1.99, Revision 2 (Reference 4), which is the basis of 10CFR50.61 (Reference 10) and used in all the analyses described herein, stipulates that these equations are valid only in the temperature range of 525° to 590°F. Therefore, the reactor vessel inlet temperature must be maintained within this range to retain validity of all existing analyses.

Low-Temperature Overpressure Protection System

The LTOP limitations are evaluated to determine if any change is required due to the power uprate.

5.1.2.4 Acceptance Criteria

Surveillance Capsule Withdrawal Schedule

The proposed surveillance capsule withdrawal schedules developed for Kewaunee following the uprating must meet the requirements of ASTM E185-82 (Reference 7). A satisfactory number of surveillance capsules will remain in the reactor vessel so that further analysis, such as for life extension, can be completed as necessary.

Heatup and Cooldown Pressure-Temperature Limit Curves

If the fluence projections for Kewaunee increase from that used in the current heatup and cooldown curves, then new applicability dates must be calculated, or new heatup and cooldown curves must be generated.

Pressurized Thermal Shock

The uprated RT_{PTS} values for all beltline materials will not exceed the screening criteria of the PTS rule. Specifically, the RT_{PTS} values of the base metal (plates or forgings) will not exceed 270°F, while the girth weld metal RT_{PTS} values will not exceed 300°F through the EOL (33 EFPY) and EOLE (51 EFPY).

Emergency Response Guideline Limits

The ERG limits will be developed to establish guidelines for operator action in the event of an emergency, such as a PTS event. The ERG categories are presented in Table 5.2-4. The category that Kewaunee must follow will be presented in subsection 5.1.2.5.

Upper Shelf Energy

At power uprate conditions, the EOL USE values for all reactor beltline materials will meet the requirements of 10CFR50, Appendix G (Reference 13).

Inlet Temperature

The inlet temperature must be maintained in the range of 525° to 590°F for the current analyses described herein to remain valid.

Low-Temperature Overpressure Protection System

The LTOPS setpoints protect the reactor vessel, so there are no violations of the reactor vessel Appendix G pressure-temperature limits.

5.1.2.5 Results

The impact of uprating was evaluated based on the current reactor vessel integrity evaluation (Reference 14). Per Section 7.5 of this document, the neutron fluence projections for Kewaunee after the Power Uprating have increased from previous analyses.

Surveillance Capsule Withdrawal Schedule

The revised fluence projections considering the Power Uprate Program have exceeded the fluence projections used in the development of the current withdrawal schedule for Kewaunee (Reference 8). A calculation of ΔRT_{NDT} at 33 EFPY was performed to determine if the increased fluences alter the number of capsules to be withdrawn for Kewaunee. This calculation determined that the maximum ΔRT_{NDT} using the uprated fluences for Kewaunee at EOL is greater than 200°F. Per Section 12 of Volume 12.02 of the *Annual Book of ASTM Standards* (Reference 7), these ΔRT_{NDT} values would require five capsules to be withdrawn from Kewaunee. The number of capsules has not changed from the current withdrawal schedule. However, due to changes in capsule fluences, the schedule has been updated as shown in Table 5.1-2. It should also be noted that the Kewaunee withdrawal schedule is acceptable since it meets the requirements of ASTM E185-82 (Reference 7).

Applicability of Heatup and Cooldown Pressure-Temperature Limit Curves

Kewaunee is currently operating to P-T limit curves documented in WCAP-14278 (Reference 9). A review based on uprated fluences was conducted on the current heatup and cooldown curve applicability date for Kewaunee. This review indicates that the revised ART after the Power Uprate Program will be more restrictive than that used in developing the current ART values for

Kewaunee at 33 EFPY. Therefore, a change in applicability date is required. The 33 EFPY P-T curves for Kewaunee will be applicable to 31.1 EFPY after the uprating, which is projected to be reached during cycle 31, on approximately April 23, 2011.

Pressurized Thermal Shock

The PTS calculations were performed for the Kewaunee intermediate and lower shell forgings using the latest procedures required by the NRC (Reference 10) and using master curve technologies for the circumferential weld. The calculated neutron fluence values for the Power Uprate Program condition at Kewaunee have increased over the current fluences (Reference 14). Therefore, to evaluate the effects of the Power Uprate Program, the PTS values for the beltline region materials from Kewaunee were re-evaluated using the uprated fluences. Based on this evaluation, all RT_{PTS} values will remain below the NRC screening criteria values using the projected Power Uprate Program fluence values through EOL (33 EFPY) and EOLE (51 EFPY) for Kewaunee, as shown in Tables 5.1-3 and 5.1-4.

Emergency Response Guideline Limits

The current peak-inside surface RT_{NDT} values at EOL that were calculated in Tables 5.1-3 and 5.1-4, are above 250°F for Kewaunee. The limiting material for Kewaunee was the circumferential weld 1P3571. The RT_{NDT} values prior to the Power Uprate Program, which are documented in WCAP-14280 (Reference 14), put Kewaunee in ERG Category IIIb per Table 5.1-5. Even though the revised fluence projections after the Power Uprate Program have increased over the fluence projections used in development of the current peak inside surface RT_{NDT} values at EOL (33 EFPY), Kewaunee will remain in ERG Category IIIb.

Upper Shelf Energy

All beltline materials are expected to have a USE greater than 50 ft-lb through the EOL (33 EFPY), as required by 10CFR50 (Reference 13). The EOL USE was predicted using the EOL 1/4T fluence projection.

The revised fluence projections associated with the Power Uprate Program have increased the fluence projections used in developing the current predicted EOL USE values (Reference 8). However, it has only affected the 1/4T fluence by 1.074 percent. This 1.07-percent increase has a slight measurable effect on the percent decrease in USE. Therefore, the current predicted

USE values for Kewaunee have been updated as shown in Table 5.1-6. Note that all USE values for Kewaunee will maintain a level above the 50 ft-lb screening criterion at EOL (33 EFPY).

Inlet Temperature

Per Section 2, which contains the new parameters, the inlet temperature for Cases 1 and 2 are below 525°F, while Cases 3 and 4 are within the range of 525° to 590°F. However, it should be noted that Kewaunee will actually operate the plant, so that these parameters are above 525°F in accordance with the acceptance criteria.

Low-Temperature Overpressure Protection System

The uprating will not be changing the actual Appendix G pressure versus temperature limits for which the LTOPS must provide protection. However, the maximum time in plant lifetime for which these pressure versus temperature limits are applicable will be changed from 33.0 EFPY to 31.1 EFPY. Therefore, since the actual Appendix G pressure versus temperature limits are not changing, the existing LTOPS setpoints will remain applicable for the Power Uprating. They will remain applicable until the actual pressure versus temperature limits (not the maximum EFPY for which they are applicable) require revision.

5.1.2.6 Conclusions

The fluence projections associated with the Power Uprate Program, while considering actual power distributions incorporated to date, will exceed the current fluence projections used in the evaluations of record (withdrawal schedules, ERG category, PTS, LTOP limitations, and USE). The effect of the higher fluence values is minimal for PTS and has not changed the ERG limits. As for the withdrawal schedule and predicted EOL USE, the effect of the higher fluences is shown in Tables 5.1-2 and 5.1-6, respectively. With respect to the P-T curves, the current curves are documented in WCAP-14278 (Reference 9). These P-T curves used fluences that were developed without the current Power Uprate Program. The new applicability date for the Kewaunee P-T curves is now 31.1 EFPY, which is projected to be reached during cycle 31 on approximately April 23, 2011. The LTOP limitations are tied directly to the P-T curves and are therefore valid through 31.1 EFPY. Lastly, the inlet temperature will be maintained above 525°F for normal operation.

It is concluded that the Kewaunee Power Uprate Program will not have significant effect on the reactor vessel integrity.

5.1.3 References

1. *ASME Boiler and Pressure Vessel Code,* Section III, Nuclear Power Plant Components, 1968 Edition with Addenda through the Winter 1968, American Society of Mechanical Engineers, New York.

2. WCAP-15345, *Addendum to the Analytical Report for the Kewaunee Nuclear Power Plant Reactor Vessel (Replacement Steam Generator Evaluation)*, December 1999.

3. Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.*

4. Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials*, Rev. 2, May 1988.

5. ASTM E185-70, *Recommended Practice for Surveillance Tests on Nuclear Reactor Vessels.*

6. WCAP-8107, *Wisconsin Public Service Corp. Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program*, S. E. Yanichko, et al, April 1973.

7. ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," *Annual Book of ASTM Standards*, Section 12, Volume 12.02.

8. WCAP-14279, *Analysis of Capsule S From the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program*, E. Terek, G. N. Wrights, and J. F. Williams, March 1995.

9. WCAP-14278, *Kewaunee Heatup and Cooldown Limit Curves for Normal Operation*, T. Laubham and C. Kim, Rev. 1, September 1998.

10. 10CFR50.61, *Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events*, Federal Register, Volume 60, No. 243, December 19, 1995.

11. WCAP-15075, *Master Curve Strategies for RPV Assessment*, R. G. Lott, M. T. Kirk, and C. C. Kim, September 1998.

12. Letter K-01-027 from NRC to Mark Reddemann, *Kewaunee Nuclear Power Plant - Request for Exemption from the Requirements of 10 CFR Part 50, Appendix G and H, and 10 CFR 50.61 (TAC No. MA8585),* dated February 21, 2001.

13. 10CFR50, Appendices G and H, *Reactor Vessel Material Surveillance Program Requirements*, Federal Register, Volume 60, No. 243, December 19, 1995.

14. WCAP-14280, *Evaluation of Pressurized Thermal Shock for the Kewaunee Reactor Vessel*, E. Terek, R. Lott, and C. Kim, Rev. 1, September 1998.

Table 5.1-1

Stress Intensities and Fatigue Usage Factors for the Kewaunee Reactor Vessel as Evaluated at 1780 MWt

Location	**Maximum Range of Primary-plus-Secondary Stress Intensity ($P_L + P_b + Q$)**	**Maximum Cumulative Fatigue Usage Factor (U_c)**
Outlet Nozzles	Nozzle: [ ]^{a,c} ksi < 3 S_m = 80.1 ksi Safe End: [ ]^{a,c} ksi < 3 S_m = 49.2 ksi	Nozzle: [ ]^{a,c} < 1.0
Inlet Nozzles	Nozzle: [ ]^{a,c} ksi < 3 S_m = 80.1 ksi Safe End: [ ]^{a,c} ksi < 3 S_m = 52.9 ksi	Nozzle: [ ]^{a,c} < 1.0
Main Closure Flange Region		
Closure Head Flange	[ ]^{a,c} ksi < 3 S_m = 80.1 ksi	[ ]^{a,c} < 1.0
Vessel Flange	[ ]^{a,c} ksi < 3 S_m = 80.1 ksi	[ ]^{a,c} < 1.0
Closure Studs	[ ]^{a,c} ksi < 3 S_m = 118.8 ksi	[ ]^{a,c} < 1.0
Vessel Shell		
Vessel Wall Transition	[ ]^{a,c} ksi < 3 S_m = 80.1 ksi	[ ]^{a,c} < 1.0
Bottom Head to Shell Juncture	[ ]^{a,c} ksi < 3 S_m = 80.1 ksi	[ ]^{a,c} < 1.0
Core Support Guides	[ ]^{a,c} ksi < 3 S_m = 69.9 ksi	[ ]^{a,c} < 1.0
CRDM Housings	[ ]^{a,c} ksi < 3 S_m = 69.9 ksi	[ ]^{a,c} < 1.0
Bottom Head Instrumentation Tubes	[ ]^{a,c} ksi < 3 S_m = 69.9 ksi	[ ]^{a,c} < 1.0
Safety Injection Nozzles	[ ]^{a,c} ksi < 3 S_m = 80.1 ksi	[ ]^{a,c} < 1.0
External Support Brackets	[ ]^{a,c} ksi < 3 S_m = 80.1 ksi	[ ]^{a,c} < 1.0

Bracketed [ ]^{a,c} information designates data that is Westinghouse Proprietary, as dicussed in Section 1.7 of this report.

Table 5.1-2 Recommended Surveillance Capsule Withdrawal Schedule				
Capsule	**Capsule Location**	**Lead Factor**	**Withdrawal EFPY[1]**	**Fluence (n/cm^2)**
V	77°	3.03	1.3	5.86×10^{18}
R	257°	3.03	4.6	1.76×10^{19}
P	247°	2.00	11.1	2.61×10^{19}
S	57°	2.08	16.2	3.67×10^{19}
T	237°	2.18	EOL	See note 2
N	67°	2.13	Standby	- - -

Notes:

1. EFPYs from plant startup.
2. Capsule T should be removed before it receives a fluence of 7.12×10^{19} n/cm^2(E>1.0 MeV) (i.e., twice the peak vessel EOL surface fluence of 3.56×10^{19} n/cm^2 (E>1.0 MeV)). This capsule may be held without testing following withdrawal. Capsule T will reach a fluence of approximately 5.63×10^{19} n/cm^2 (E> 1.0 MeV) at 22.23 EFPY, which was reached during cycle 25 on approximately February 26, 2002. This is equal to the reactor vessel peak surface fluence of 5.63×10^{19} n/cm^2(E>1.0 MeV) at 51 EFPY (approximately December 31, 2033, during fuel cycle 46, 60 calendar year life).

Table 5.1-3

RT_{PTS} Calculations for Kewaunee Intermediate and Lower Shell Materials at 33 EFPY and 51 EFPY with Uprated Fluences Using Charpy-Based Data

Material	Fluence (n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS} (3) (°F)	Margin (°F)	$RT_{NDT(U)}$ (1) (°F)	RT_{PTS} (2) (°F)
33 EFPY							
Intermediate Shell Forging 122X208VA1	3.56	1.33	37	49.2	34	60	143
→ Using S/C Data	3.56	1.33	23.8	31.6	34	60	126
Lower Shell Forging 123X167VA1	3.56	1.33	37	49.2	34	20	103
→ Using S/C Data	3.56	1.33	21.2	28.2	34	20	82
51 EFPY							
Intermediate Shell Forging 122X208VA1	5.63	1.425	37	52.7	34	60	147
→ Using S/C Data	5.63	1.425	23.8	33.9	34	60	128
Lower Shell Forging 123X167VA1	5.63	1.425	37	52.7	34	20	107
→ Using S/C Data	5.63	1.425	21.2	30.2	34	20	84

Notes:

1. Initial RT_{NDT} values are measured values.
2. $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)
3. ΔRT_{PTS} = CF * FF

Table 5.1-4 RT_{PTS} Calculations for Kewaunee Circumferential Weld 1P3571 at 33 EFPY and 51 EFPY with Uprated Fluences Using Master Curve Technology							
Material	Fluence (n/cm^2, E>1.0 MeV)	FF	T_o (°F)	$RT_o = T_o + 33°F$ (2)	Margin (1) (°F)	Bias (1) (°F)	RT_{PTS} (°F)
33 EFPY							
Circumferential Weld 1P3571	3.56	1.330	167.5	200.5	62.5	8.5	271.5
51 EFPY							
Circumferential Weld 1P3571	5.63	1.425	190.5	223.5	62.5	8.5	294.5

Notes:

1. Per NRC in Reference 8, margin term is 62.5°F and bias term is 8.5°F.
2. Per NRC in Reference 8, $RT_o = T_o + 33°F$.

Table 5.1-5 ERG Pressure-Temperature Limits (Reference 6)	
Applicable RT_{NDT} (ART) Value (1)	ERG Pressure-Temperature Limit Category
$RT_{NDT} < 200°F$	Category I
$200°F < RT_{NDT} < 250°F$	Category II
$250°F < RT_{NDT} < 300°F$	Category IIIb

Note:

1. Longitudinally oriented flaws are applicable only up to 250°F; the circumferentially oriented flaws are applicable up to 300°F

Table 5.1-6 Predicted EOL (33 EFPY) USE Calculations for all the Beltline Region Materials					
Material	**Weight % of Cu**	**1/4T EOL Fluence (10^{19} n/cm^2)**	**Unirradiated USE (ft-lb)**	**Projected USE Decrease[1] (%)**	**Projected EOL USE (ft-lb)**
Intermediate Shell Forging 122X208VA1	0.06	2.41	92	7.5	85
Lower Shell Forging 123X167VA1	0.06	2.41	97	3.5	94
Circumferential Weld 1P3571	0.287	2.41	126	46	68

Note:

1. Values are deduced from Figure 2 of Regulatory Guide 1.99, Revision 2.

5.2 Reactor Pressure Vessel System for Kewaunee

5.2.1 Introduction and Background

The reactor pressure vessel (RPV) system consists of the reactor vessel, reactor internals, fuel, and control rod drive mechanisms (CRDMs). The reactor internals 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 direct coolant flow through the fuel assemblies, provide adequate cooling flow to the various internal structures, and support in-core instrumentation. They are designed to withstand forces due to structural deadweight, preload of fuel assemblies, control rod assembly dynamic loads, vibratory loads, earthquake accelerations, and loss-of-coolant accident (LOCA) loads.

Evaluation of the uprated conditions requires that the reactor vessel/internals/fuel system interface be assessed to ensure 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 for input to LOCA and non-LOCA safety analyses and Nuclear Steam Supply System (NSSS) performance evaluations.

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

- Reactor internals system thermal-hydraulic performance
- Rod control cluster assembly (RCCA) scram performance
- Reactor internals system structural response and integrity

5.2.2 Input Parameters and Assumptions

5.2.2.1 Operating Parameters

The operating parameters (pressure, temperature, flow, power level) are per the Performance Capability Working Group (PCWG) parameters in Section 2 of this report. Also, the design transients (Section 3) were used in the evaluation.

5.2.2.2 Fuel

A full core of Westinghouse 422 V+ fuel is assumed. The LOCA forces and resultant loads on the fuel are valid for either Westinghouse fuel or ANF fuel, so mixed cores are adequately addressed.

5.2.3 Description of Analyses and Evaluations

Descriptions of the various analyses and evaluations are given in the individual subsections of 5.2.5 through 5.2.7.

5.2.4 Acceptance Criteria

Acceptance criteria are typically listed in each individual section. The most important acceptance criteria are grouped together below.

5.2.4.1 Core Bypass Flow

The core bypass flow must be within the 7.0-percent upper limit.

5.2.4.2 Hydraulic Forces

Hydraulic forces on the reactor internal must be limited so that the internal will remain seated and stable.

5.2.4.3 Cumulative Fatigue Usage Factor

The cumulative fatigue usage factor for the most critically stressed components will be less than 1.0.

5.2.5 Thermal-Hydraulic System Evaluations

5.2.5.1 System Pressure Losses

The principal Reactor Coolant System (RCS) flow route through the RPV system at the Kewaunee unit begins at the two inlet nozzles. At this point, flow turns downward through the reactor vessel core barrel annulus. After passing through this downcomer region, 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 two outlet nozzles. Note that support columns and RCCA guide columns occupy the upper plenum region.

A key area in evaluating core performance is to determine the hydraulic behavior of coolant flow within the reactor internals system, that is, vessel pressure drops, core bypass flows, RPV fluid temperatures and hydraulic lift forces. The analyses determined the distribution of pressure and flow within the reactor vessel, internals, and the reactor core for the uprated conditions.

5.2.5.2 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 the bypass flow paths occur during manufacturing, such as gaps at the outlet nozzles and core barrel, or changes due to different fuel assembly designs or changes in RCS conditions, plant-specific as-built dimensions are used to demonstrate that bypass flow limits are not violated. Analyses are performed to determine core bypass flow values. This ensures that either the design bypass flow limit for the plant will not be exceeded or a revised design core bypass flow is required. If required, a revised design core bypass flow is developed, and core bypass flow values are re-analyzed.

The present design core bypass flow limit is 7.0 percent (with thimble plugs removed) of the total reactor vessel flow. The purpose of this evaluation is to ensure that the design value of 7.0 percent can be maintained at the uprated RCS conditions. The principal core bypass flow paths are the:

- Baffle-barrel region
- Vessel head cooling spray nozzles
- Core barrel – reactor vessel outlet nozzle gap
- Fuel assembly – baffle plate cavity gap
- Fuel assembly thimble tubes

Fuel assembly hydraulic characteristics and system parameters, such as reactor coolant inlet temperature, pressure, and flows were used to determine the impact of uprated conditions on total core bypass flow. The design core bypass flow value of 7.0 percent of the total vessel flow can be maintained at the uprated conditions.

5.2.5.3 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 and upper internals flanges, and the reactor vessel shell flange and core barrel flange of the internals.

An evaluation was performed for the 7.4-percent uprated conditions to determine the effects of hydraulic lift forces on the various reactor internal components. The results show that the uplift forces are less than previously analyzed and are therefore acceptable.

5.2.5.4 Rod Control Cluster Assembly Scram Performance Evaluation

The RCCAs represent the interface between the fuel assemblies and other internal components. An evaluation was performed to determine the potential impact due to power uprating at Kewaunee on RCCA scram characteristics used in the *Updated Safety Analysis Report* (USAR) accident analyses. This analysis is based on the Westinghouse 422 V+ fuel assemblies.

The evaluation indicated that, for even the most severe case, the current limit for drop-time-to-dashpot entry of 1.8 seconds remains applicable for accident analyses.

5.2.5.5 Momentum Flux and Fuel Rod Stability

Baffle jetting is a hydraulically induced instability or vibration of fuel rods caused by a high-velocity jet of water. This jet is created by high-pressure water being forced through gaps between the baffle plates surrounding the core. The baffle jetting could lead to fuel-cladding damage.

To assess the impact of the uprated RCS conditions on baffle jetting margins of safety at the Kewaunee unit, the ratio of the margins of safety between the present plant configuration and the uprated configuration has been determined. The results show that, based on mechanical design flow, the margins of safety for momentum flux at uprated conditions do not change significantly from those at the present conditions.

5.2.6 Mechanical System Evaluations

The RCS mechanical response, 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, which would result from release of equilibrium forces at the break locations, are obtained. In the second step, the loop mechanical loads and 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 (RCL) and component supports to deal with the LOCA are evaluated by applying the calculated reactor vessel displacements to a mathematical model of the RCL. Thus, the effects of vessel displacements upon the loop and reactor vessel/internals are evaluated.

5.2.6.1 Loss-of-Coolant Accident Loads

LOCA loads applied to the Kewaunee RPV system consist of reactor internal hydraulic loads (vertical and horizontal), and pressure loads acting on the baffle plates. All loads are calculated individually and combined in a time-history manner.

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.

Since the Kewaunee reactor takes credit for leak-before-break (LBB) applied to the primary loop, LOCA analyses of the RPV system for postulated ruptures of primary loop piping are not required. The next limiting breaks to be considered are branch-line breaks, such as in the accumulator line, pressurizer surge line, and residual heat removal (RHR) line. With consideration of LBB on the primary lines, such auxiliary line breaks are not as severe as the main line breaks (for example, RPV inlet nozzle or RCP outlet nozzle break).

The LOCA forces generated for Kewaunee for these branch line breaks are much lower than those originally considered for the reactor pressure vessel inlet nozzle and reactor coolant pump outlet nozzle breaks. Therefore, critical stresses in baffle and former plates and associated bolting are bounded by existing analyses (Reference 1).

5.2.6.2 Flow-Induced Vibrations

FIVs of pressurized water reactor internals have been studied at Westinghouse for a number of years. The objective of these studies was to ensure the structural integrity and reliability of reactor internal components.

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

The PCWG parameters, which could potentially influence FIV response of the reactor internals, include inlet nozzle flow velocities, vessel/core inlet temperatures, and vessel outlet temperatures. Generally, the inlet nozzle velocity for FIV response during hot functional testing is calculated using mechanical design flows, which are approximately 15 percent higher than thermal design flows.

The other parameter, which would influence the FIV response, is core inlet temperature. For the most limiting case at uprated conditions, the vessel/core inlet temperature is 521.9°F. For the uprated conditions, it was determined that FIV loads on the guide tubes, upper support columns, and the lower internals are within 5 percent of previous FIV analyses. There is sufficient margin to accommodate this increase in FIV loads. Consequently, the structural integrity of the Kewaunee reactor internals remains acceptable with regard to FIVs.

5.2.7 Structural Evaluation of Reactor Internal Components

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 primary flow through the core, the internals assembly also establishes secondary flow paths for cooling the upper regions of the reactor vessel and 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 flow rate required to remove heat generated within the structural components due to radiation (for example, gamma heating). The configuration of the internals provides 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 demonstrate that the structural integrity of reactor internal components is not adversely affected either directly by the uprated RCS conditions and transients, or by secondary effects on reactor thermal-hydraulic or structural performance. Heat generated in reactor internal components, along with the various fluid temperature changes, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth, which must be considered in the design and analysis of the various components.

Since the Kewaunee reactor internals were designed prior to 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 Kewaunee reactor internals was evaluated according to Westinghouse internal criteria, which is similar to the ASME Code (Reference 2). Moreover, the structural integrity of the Kewaunee reactor internals design has been ensured by analyses performed on both generic and plant-specific bases. These analyses were used as the basis for evaluating critical Kewaunee reactor internal components for uprating and revised design transients.

5.2.7.1 Lower Core Plate

Structural evaluations were performed to demonstrate that the structural integrity of the lower core plate is not adversely affected either by the uprated RCS conditions or by secondary effects 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 were utilized.

Primarily because of the higher gamma heating rates associated with the uprated conditions, the lower core plate is the most critically stressed component in the entire reactor internals assembly. The conclusion of these evaluations is that structural integrity of the lower core plate is maintained. The uprated RCS conditions produced acceptable margins of safety and fatigue utilization factors for all ligaments under all loading conditions. The limiting (highest) cumulative usage factor for the Kewaunee lower core plate at the 7.4-percent power uprate is [ ]a,c, whereas 1.0 is the ASME Code limit.

5.2.7.2 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 exiting the fuel assemblies and serves as a boundary between the core and exit plenum. The upper core plate is restrained from vertical movement by the upper support columns, which are attached to the upper support plate assembly. Four equally spaced core plate alignment pins restrain lateral movement.

The stresses in the upper core plate are mainly due to hydraulic and thermal loads. The total thermal stresses are due to thermal bending moments through the thickness and surface peak stresses. An evaluation was performed to determine the impact of uprating 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 uprated RCS conditions at the Kewaunee unit. The limiting (highest) cumulative usage factor for the Kewaunee upper core plate at 7.4-percent power uprate is [ ]a,c, whereas 1.0 is the ASME Code limit.

5.2.7.3 Baffle-Barrel Region Components

The Kewaunee lower internals assembly consists of a core barrel into which baffle plates are installed, supported by interconnecting former plates. A lower core support structure is provided at the bottom of the core barrel and a neutron panel surrounds the core barrel. The components comprising the lower internals assembly are precision-machined. The baffle and former plates are bolted into the core barrel. The reactor vessel internals configuration for Kewaunee utilizes downflow in the barrel-baffle region.

5.2.7.3.1 Core Barrel Evaluation

The thermal stresses in the core active region of the core barrel shell are primarily due to temperature gradients through the thickness of the core barrel shell. Evaluations were performed to determine the thermal bending and skin stresses in the core barrel for the uprated RCS conditions. These evaluations indicated that the actual number of fatigue cycles, based on all normal/upset conditions, was well below the allowable value. From these conservative results, it has been concluded that the core barrel is structurally adequate for the Kewaunee uprated RCS conditions.

5.2.7.3.2 Baffle-Barrel Bolt Evaluation

The bolts are evaluated for loads resulting from hydraulic pressure, seismic loads, preload, and thermal conditions. The temperature difference between baffle and barrel produces the dominant loads on the baffle-former bolts. Hydraulic pressure and seismic loads produce the primary stresses, whereas bolt preloading and thermal conditions produce the secondary stresses. The uprated RCS conditions do not affect deadweight or preload forces.

Since these bolts are qualified by test, the evaluation of the revised loads consisted of comparing the existing operating loads to those developed with the uprated RCS conditions. The results indicate that the thermal baffle-former and barrel-former bolt loads, with the currently analyzed condition, bound those developed with the uprated RCS conditions. Therefore, it is concluded that the baffle-former and barrel-former bolts are structurally adequate for the uprated RCS conditions.

5.2.7.4 Additional Components

A series of assessments were performed on reactor internal components that were not significantly impacted by the power uprating (and the resulting internal heat generation rates), but are affected by the uprated RCS conditions due to primary loop design transients. These components are:

- Lower core support plate
- Lower support columns
- Core barrel outlet nozzle
- Core barrel flange
- Lower radial restraints (clevis inserts)
- Upper core plate alignment pin
- Upper support columns

- Upper support plate
- Guide tubes and support pins

The results of these assessments demonstrate that the above listed components are structurally adequate for the uprated RCS conditions.

5.2.8 Conclusions

Analyses have been performed to assess the effect of changes due to power uprate. The results of these analyses are as follows:

- The design core bypass flow value of 7.0 percent of the total vessel flow is maintained for the uprating.

- Hydraulic forces at the uprated conditions were evaluated for effects on the reactor internals. It was determined that the Kewaunee reactor internals assemblies will remain seated and stable at the uprated conditions.

- An RCCA performance evaluation was completed and indicated that the current 1.8-second RCCA drop-time-to-dashpot entry limit (from gripper release of the drive rod) is satisfied at power-uprate conditions.

- Baffle plate momentum flux margins of safety due to power-uprate conditions are relatively unchanged from present conditions for mechanical design flow, and remain acceptable.

- Evaluations were completed and indicated that the uprated RCS conditions will not adversely impact the response of reactor internals systems and components due to seismic/LOCA excitations and FIVs.

- Evaluations of the critical reactor internal components were performed, which indicated that the structural integrity of the reactor internals is maintained at the uprated RCS conditions. Limiting cumulative usage factors are shown in Table 5.2-1.

5.2.9 References

1. WCAP-9958, *Generic Stress Report for Two-Loop Core Support Structures* (Proprietary), original issue.

2. *ASME Code Section III*, Appendices, 1989 Edition (used as a guideline; there is no code of record for the Kewaunee reactor internals).

Table 5.2-1 Limiting Cumulative Usage Factors	
Reactor Internal Component	**CUF**
Lower Core Plate	[ ]a,c
Upper Core Plate	[ ]a,c

Bracketed [ ]a,c information designates data that is Westinghouse Proprietary, as dicussed in Section 1.7 of this report.