NDP-98-181 (REVISION 1)

TRITIUM PRODUCTION CORE (TPC) TOPICAL REPORT

(UNCLASSIFIED, NON-PROPRIETARY VERSION)

(February 8, 1999)

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LIST OF ACRONYMS

AFD	Axial Flux Difference
ALARA	As Low As Reasonably Achievable
ALI	Annual Limit on Intake
AMSAC	ATWS Mitigation System Actuation Circuitry
ANS	American Nuclear Society
a/o	atom percent
ARI	All Rods In
ARO	All Rods Out
ART	Adjusted Reference Temperature
ATR	Advanced Test Reactor
ATWS	Anticipated Transients Without Scram
BA	Burnable Absorber
BISI	Bypassed and Inoperable Status Indication
BMTC	Bottom Mounted Thermocouples
BOL	Beginning of Life
BOP	Balance of Plant
BP	Burnable Poison
BPRA	Burnable Poison Rod Assembly (WABA and/or Pyrex Burnable Absorber)
CFR	Code of Federal Regulations
CL	Centerline of Fuel Stack
CLWR	Commercial Light Water Reactor
COLR	Core Operating Limits Report
COMS	Cold Overpressure Mitigation System
CRD	Confidential Restricted Data
CRDM	Control Rod Drive Mechanism
CRDS	Control Rod Drive Systems
CRSD	Core Radiation Source Data
CVCS	Chemical and Volume Control System
CZP	Cold Zero Power
DAC	Derived Air Concentration
DBA	Design Basis Accident
DNB	Departure from Nucleate Boiling

LIST OF ACRONYMS (Cont.)

DNBR	Departure from Nucleate Boiling Ratio
DOE	Department of Energy
EAS	Essential Auxiliary Support
EC	Equilibrium Core
ECCS	Emergency Core Cooling System
EFPD	Effective Full Power Days
EFPY	Effective Full Power Years
EOL	End of Life
EQ	Environmental Qualification
ERG	Emergency Response Guideline
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
FC	First Core
FIV	Flow Induced Vibration
FON	Fraction of Nominal
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GVR	Gas Volume Ratio
HFP	Hot Full Power
H/M	Hydrogen - to - Metal Ratio
HZP	Hot Zero Power
ID	Inner Diameter
IFBA	Integral Fuel Burnable Absorber
IFM	Intermediate Flow Mixer
LAR	License Amendment Request
LBLOCA	Large Break Loss of Coolant Accident
LOCA	Loss of Coolant Accident
LTA	Lead Test Assembly
LTOPS	Low Temperature, Overpressure Protection System
LWNPR	Light Water New Production Reactor
M/E	Mass and Energy
M&TE	Measurement and Test Equipment

LIST OF ACRONYMS (Cont.)

MPH	Material Properties Handbook
MTC	Moderator Temperature Coefficient
MWD/MTU	Megawatt Days per Metric Ton of Uranium
MWt	Megawatt Thermal
NDE	Nondestructive Examination
NIS	Nuclear Instrumentation System
NIST	National Institute of Standards and Technology
NPSH	Net Positive Suction Head
NPZ	Nickel-Plated Zirconium
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OD	Outer Diameter
ODCM	Offsite Dose Calculation Manual
OPΔT	Overpower Delta Temperature
ORE	Occupational Radiation Exposure
OTΔT	Overtemperature Delta Temperature
PCT	Peak Cladding Temperature
PIE	Post-Irradiation Examination
PMA	Process Measurement Accuracy
PNNL	Pacific Northwest National Laboratory
PORV	Power Operated Relief Valve
PRF	Permeation Reduction Factor
P-T	Pressure-Temperature
PTLR	Pressure Temperature Limits Report
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
QMS	Quality Management System
RAOC	Relaxed Axial Offset Control
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump

LIST OF ACRONYMS (Cont.)

RCS	Reactor Coolant System
RHRS	Residual Heat Removal System
RPVSA	Reactor Pressure Vessel System Analysis
RSD	Relative Standard Deviation
RT	Reference Temperature
RTDP	Revised Thermal Design Procedure
RTP	Rated Thermal Power
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
SAL	Safety Analysis Limit
SBLOCA	Small Break Loss of Coolant Accident
SCC	Stress Corrosion Cracking
SDM	Shutdown Margin
SER	Safety Evaluation Report
SFP	Spent Fuel Pit
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SRP	Standard Review Plan
SS	Stainless Steel
SSE	Safe Shutdown Earthquake
STDP	Standard Thermal Design Procedure

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UCP	Upper Core Plate
UET	Unfavorable Exposure Time
USE	Upper Shelf Energy
VAMT	Vessel Average Moderator Temperature
WABA	Wet Annular Burnable Absorber

LIST OF ACRONYMS (Cont.)

WAES	Westinghouse Advanced Energy Systems
WBNP	Watts Bar Nuclear Plant
w/o	Weight Percent or Without (depending on context)
WOG	Westinghouse Owners Group

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ABSTRACT

The U.S. Department of Energy plans to retain the option to produce tritium in one or more commercial light water reactors (CLWRs) with tritium producing burnable absorber rods (TPBARs). This topical report describes how the inclusion of significant numbers of TPBARs affects nuclear plant systems, safety and components analyses and performance for a representative CLWR. The intent of this report is to serve as a guide and referenceable document for plant-specific efforts to license incorporation of TPBARs for any CLWR design in the United States. For the phase of effort described in this topical report, evaluations and analyses were performed assuming a conservatively large number of TPBARs in the core (> 3000), and core designs (first and equilibrium) that are not optimized for fuel usage. A reference CLWR design was used to provide a representative configuration upon which the impact of the full core of TPBARs was determined. The results of this effort are not bounding for all CLWRs, but bounding for many characteristics and representative of the effort and anticipated outcome. It is expected that plant-specific efforts will be necessary in the nuclear steam supply system (NSSS) safety, systems, and components analysis areas, but the nature and extent of that effort are outlined in this topical report.

The NRC Standard Review Plan (NUREG-0800) was used as a guide to determine the areas for evaluation in the NSSS. The Lead Test Assembly submittal and related licensing documentation serve as the reference and basis for further licensing evaluation of the TPBARs reflected in this topical report. The results of the evaluation for the representative CLWR demonstrate that operation of a CLWR with a large number of TPBARs in the core is feasible and does not have a significant adverse impact on the safety, operability, and productivity of the plant.

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SECTION 1 INTRODUCTION

1.1 PURPOSE OF PROGRAM

The U.S. Department of Energy (DOE) is planning to produce tritium for the National Security Stockpile. One source for the production of tritium is the irradiation of Tritium Producing Burnable Absorber Rods (TPBARs) in a commercial light water reactor. Figure 1-1 shows how the TPBARs are arranged and installed into the fuel assemblies. This topical report addresses the safety and licensing issues associated with incorporating a full complement of TPBARs in a commercial light water reactor (CLWR), specifically a pressurized water reactor (PWR). This report is submitted to the NRC for review, with the request that the NRC issue a Safety Evaluation Report (SER) to support the licensing of the use of TPBARs in PWRs. In addition to demonstrating that no significant safety issues are raised by the operation of a reference plant with a full complement of TPBARs, this Topical Report provides a methodology and a reference for a utility to use in a plant specific evaluation. This topical report demonstrates that a licensing basis exists for the use of a full complement by a utility in a PWR. The core design evaluated is based on maximizing the number of TPBARs in the core while limiting them to only available guide thimble tube locations within the fuel assemblies. Optimization of the fuel cycle for fuel usage may be performed to the extent desired for a plant specific application.

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1.2 DESCRIPTION OF EFFORT

The scope of effort concentrated on:

- evaluation of the impact of TPBARs on the safety and operability of the reference PWR and
- evaluation of the integrity and performance of the TPBARs within the tritium production core (TPC).

The NRC Standard Review Plan (SRP) (NUREG-0800, Reference 1) was used as the basis for evaluating the impact of the TPBARs on the reference plant. All technical sections of the SRP are specifically addressed in this report. Section 2 of this Topical Report addresses each chapter of the SRP, and contains descriptions of the input, methodology, and results relative to the TPBAR impact on the reference reactor. The majority of Section 2 is dedicated to the portions of the SRP where there is an impact from inclusion of TPBARs; however, for the unaffected sections of the SRP, the reasons for the judgment of "no impact" are provided. This strategy was selected as the most systematic basis for evaluating the impact of the TPBARs on plant safety, which is the primary focus for this report. Core designs for first and equilibrium cores were developed, specifically for the maximum number of TPBARs. The key safety analysis parameters resulting from the TPC design were used in the safety analyses for comparison with typical reference reactor values and for use in safety evaluations. In response to SRP 4.3, Section 2 contains a description of the core designs used for the Topical Report evaluations.

Previously, DOE issued a technical report (Reference 2) for the irradiation of TPBAR Lead Test Assemblies (LTAs) in a commercial light water reactor and TVA submitted a License Amendment Request (LAR) to irradiate the TPBAR LTAs in the Watts Bar Reactor, Cycle 2. The NRC issued a license amendment for the LTAs (Reference 3), and the LTAs were installed in the Watts Bar reactor in the fall of 1997. In selecting the reference reactor for use in preparing this report, an uprated plant with dry containment was judged to be more limiting with a full complement of TPBARs, and therefore the more limiting plant was selected as the reference plant. The LTA TPBAR design was modified for the TPC to account for the differences in core design between Watts Bar Cycle 2 with 32 LTA TPBARs and a production core with more than 3300 TPBARs. The LTAs were placed in non-limiting locations in the reactor core whereas the production core TPBARs are located in every available reactor core location, which may include the limiting core fuel assemblies. In this report, the production core TPBARs are compared to the LTA TPBARs for the convenience of the reader.

Throughout this report, the following terms are used to distinguish a tritium production reactor from the LTA and CLWR reference reactors:

reference reactor or plant - A reactor plant that has no TPBARs and therefore does not purposely produce tritium. The reference reactor differs from the reactor used for the Lead Test Assemblies, but both reactors have similar characteristics.

tritium production reactor – The reference reactor or plant, with a core designed to produce tritium using a maximum complement of TPBARs.

reference core - The core (193 fuel assemblies) in the reference reactor.

tritium production core (TPC) - The reactor core having 193 fuel assemblies and the maximum complement of TPBAR assemblies.

For the evaluation of the effects of plant operating conditions on the TPBAR itself, the evaluation format used for the Lead Test Assembly (Reference 1) was used. For the LTA program, the effects of CLWR operating conditions on TPBARs were evaluated/analyzed, and the LTA was approved for installation into the Watts Bar core for Cycle 2. The focus for the Topical Report is on the reactor plant aspects affected by the greater numbers of TPBARs in the core. Section 3 of this report contains the various aspects of the TPBAR evaluation, including the design requirements, the mechanical and thermal hydraulic evaluations, the performance, nuclear design interfaces, materials considerations, test data, and surveillance.

The use of TPBARs in every available fuel assembly guide thimble location (i.e., in every location other than under control rods or in guide thimbles containing primary/secondary source rods) is considered to be limiting. Although maximizing the number of TPBARs in the core is generally bounding, in a few instances, the inclusion of TPBARs mitigates the consequences of an accident. Therefore, a LAR for a partial core loading of TPBARs will need to address those accidents where a full complement of TPBARs is not bounding. Use of fresh TPBARs in spent reactor fuel is not precluded by the core design, but may require the use of specialized handling tooling at the reactor site. Use of TPBARs in new (fresh) fuel does not require specialized tooling. The production scenario developed here assumes that the fresh fuel arrives at the reactor with the TPBARs installed and that no removal of the TPBARs from the fuel occurs before the fuel is installed into the reactor.

The scope of this report covers the period between when the fresh fuel assemblies (containing unirradiated TPBARs) arrive at the reactor site and the fuel completes one cycle and is returned to the spent fuel pool with irradiated TPBARs. The TPBARs are designed for only one reactor cycle and are removed from the fuel and prepared for shipment to a DOE facility. (In the equilibrium fuel cycle defined in Section 2.4.3, some of the once burnt fuel assemblies are reinserted for a second cycle under control rod assemblies, but without TPBARs.) In a similar manner, commercial reactors use discrete burnable absorber rods in fuel assemblies during the first cycle, then remove them from the fuel assemblies and reuse the fuel during subsequent cycles. The activities required to remove the TPBARs from the fuel assemblies and prepare them for shipment are dependent on the fuel pool design, making it difficult to generically address post-irradiation handling of TPBARs, and therefore are not included in this report. Discussion of these activities will be covered in either a separate topical report or in a LAR.

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Section 4 contains the conclusions of the evaluations and analyses performed, as well as the determination of No Significant Hazards evaluation performed for the reference plant.

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1.3 REFERENCE PLANT PARAMETERS

The reference plant evaluated for the Topical Report is a newer-vintage, four loop Pressurized Water Reactor (PWR) Nuclear Steam Supply System (NSSS), with parameters and features as provided in Table 1-1. The values provided in Table 1-1 are identical for both the reference plant and the TPC. Table 1-2 compares a few key core parameters for the reference reactor and the tritium production reactor. The production core will use TPBARs in the guide thimble locations instead of thimble plugs, for all fuel assemblies not located under control rods (except where source rods are located) (Figure 1-1). Table 1-3 provides a few key core parameters for the tritium production reactor.

This configuration is reasonably representative of candidate plants for the CLWR tritium program. It is also rated at a conservatively high power level, relative to the candidate CLWRs. In addition, for conservatism, 10% steam generator tube plugging is assumed. The NSSS performance parameters for the representative plant remain unaffected by the incorporation of the TPBARs in the core with the exception of core bypass flow, which decreases with respect to the value assumed for the reference plant. The reference plant value of 8.4% core bypass flow assumes thimble plugs removed; this is conservative for NSSS analyses, since it results in higher core average and core outlet temperatures. Therefore, it is retained for the Topical Report evaluations, because it bounds the expected value of ~6% for the full core of TPBARs.

The parameters provided in this section are primarily NSSS performance parameters. Other parameter values, such as core peaking factors, peak clad temperature, etc., are provided in Sections 2 and 3, since they are products of the analyses and evaluations performed for the Topical Report program.

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1.4 GENERIC APPLICATION OF TOPICAL REPORT

It is assumed the CLWR(s) selected to implement a TPC will be of a U.S. PWR design; the methodology and scope outlined in this report are intended to be applicable for any U.S. PWR vendor. For any CLWR selected, additional plant specific work will be necessary to obtain a license amendment, even for the reference plant. This is because the analysis margins vary from plant to plant and because fuel management and core design assumptions and objectives may differ from those employed here. As a result of parameter and configuration differences from plant to plant, as well as the potential to optimize fuel design, the extent of the detailed analysis required for each implementing plant will vary. However, it is expected that the nature of the efforts required, as described within this Topical Report, will be consistent from one CLWR to another, in spite of plant differences. This is, in part, a result of using the evaluation criteria from the Standard Review Plan. Individual Topical Report sections describe where scope differs for different configurations.

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1.5 REFERENCES

1. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition", dated June 1987, by U.S. NRC.

2. Technical Report PNNL-11419, Rev. 1, "Report on the Evaluation of the Tritium Producing Burnable Absorber Rod Lead Test Assembly", dated March 1997, by Pacific Northwest National Laboratory.

3. Letter from R. E. Martin, NRC, to O. D. Kingsley, TVA, "Issuance of Amendment on Tritium Producing Burnable Absorber Rod Lead Test Assemblies", September 15, 1997.

Table 1-1
NSSS Performance Parameters

Key Configuration Parameters:	
4 Loop Plant	
7000 Horsepower Reactor Coolant Pump	
17 x17 Fuel Assembly Rod Array	
Dry Containment	
NSSS Performance Parameters:	
NSSS Power, MWt	3579
Reactor Power, MWt	3565
Thermal Design Flow, gpm/loop	93600
Reactor Coolant Pressure, psia	2250
Core Bypass Flow Fraction	8.4%
Reactor Coolant Temperatures, °F	
Pressurizer (T_{pzr})	653.0
Core Outlet	625.0
Vessel Outlet (T_{hot})	620.0
Core Average	593.0
Vessel Average	588.4
Vessel/Core Inlet (T_{cold})	556.8
Steam Generator Outlet	556.5
Steam Generator Performance	
Steam Temperature, °F	538.4
Steam Pressure, psia	950
Steam Flow, million lb/hr	15.92
Feedwater Temperature, °F	446
SG Maximum Tube Plugging, %	10

Table 1-2 Core Design Parameters for the TPC			
Design Parameter	**Reference Plant - Typical**	**TPC First Cycle**	**TPC Equilibrium Cycle**
Total Number of Feed Assemblies	84 - 89	193	140
Feed Loading (MTU)	35.5 - 37.7	81.6	59.2
Number of TPBARs	0	3,342	3,344
Total Grams of Tritium Produced	NA	2860	2805

Table 1-3 **Key Physical Parameters**	
Fuel assemblies in the core	193
Number of RCCAs	53
Fuel rods per assembly	264
Available guide thimble tubes per assembly	24
Active length of fuel, inches	144
Active length of TPBARs, inches – First Cycle (FC) – Equilibrium Cycle (EC)	127.5 128.5
Number of fuel assemblies with source rods - FC - EC	4 4
Number of primary source rods per core - FC - EC	2 0
Number of secondary source rods per core - FC - EC	4 4

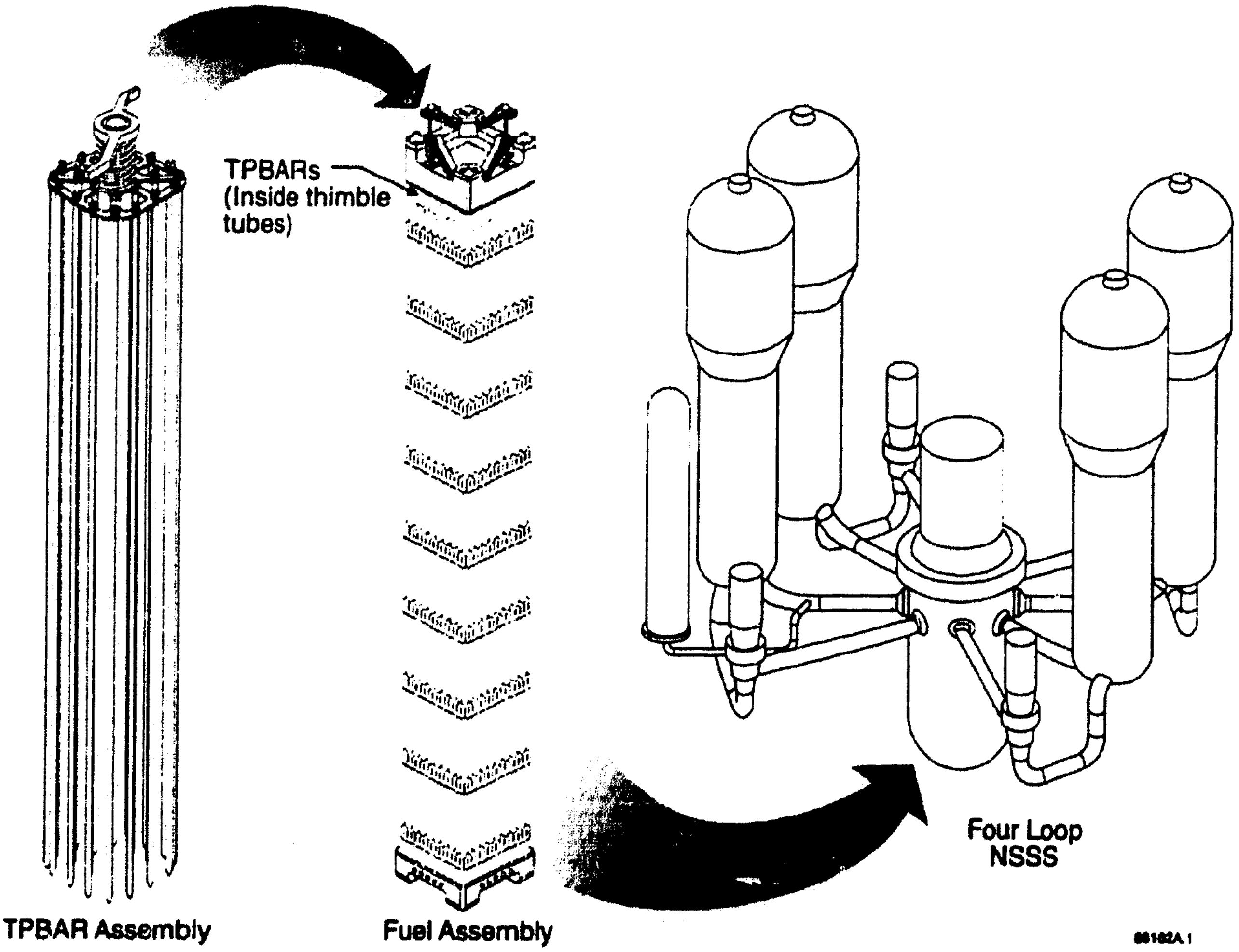

Figure 1-1
TPBAR Location in the NSSS

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SECTION 2 STANDARD REVIEW PLAN EVALUATION

This section of the report is structured to follow the format of the NRC Standard Review Plan (NUREG-0800). Each major subsection (e.g., 2.1, 2.2, etc.) is associated with a Chapter of NUREG-0800 and is titled to coincide with the corresponding chapter. The intent of this section is not to provide a complete Safety Analysis Report for the Tritium Production Core (TPC), but to provide a methodical evaluation of the impact of incorporating a full complement of TPBARs in a typical CLWR. Each major subsection begins with a brief description of all the SRPs in that chapter of NUREG-0800. For those SRPs which are clearly not impacted by the TPBARs, and no evaluation was required, the basis for that judgment is provided. Where evaluations were performed to determine the impact, a summary of those evaluations is provided.

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2.1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

Chapter 1 in a standard SAR format provides a summary description of all aspects of the plant. Since it basically consists of summaries of information found in other sections or background information, there are no explicit review plans in the SRP for most of Chapter 1. One exception is SRP 1.8, Interfaces for Standard Designs. This review plan, however, is specifically related to the review of a standard design submittal and deals with the safety-related interfaces between a standard design (whether NSSS or balance of plant, BOP) and the matching systems, components, and structures within the remaining unspecified portion of the plant design. The TPC does not involve the concept of a standard NSSS or BOP design, and any plant incorporating TPBARs will provide a License Amendment Request specific to their design. Therefore, SRP 1.8 is not applicable.

Although there are no review plans applicable to Chapter 1 of a SAR incorporating TPBARs, there is a brief description of the reactor core in SAR Section 1.2.3.1 (Reactor Core) which discusses burnable absorber rods, generically, as a means of suppressing reactivity at beginning of life. Any SAR revision to incorporate TPBARs should include a brief discussion of the TPBARs in that section. Other information in a typical SAR Chapter 1 is of such a broad nature that the incorporation of TPBARs would not change the description.

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2.2 SITE CHARACTERISTICS

The sections in Chapter 2 of the SRP deal primarily with the physical location and characteristics of the site itself, not with the fuel or core design. Each section is described briefly below, and no impacts are noted for all sections. Although SRP 2.4.13 pertains to the accidental release of liquid effluents, the consequences of which are dependent on the concentration of radioactive material (specifically tritium) assumed to be present in the spill, the increase in primary coolant discharge (see Section 2.11.3) for normal operation will assure that the tritium concentration in liquid waste spills remains within that for current operation. Therefore, it is not anticipated that there would be changes to Chapter 2 of a typical SAR as a result of incorporating TPBARs.

SRP 2.1.1
Site Location and Description: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100.10 and 10 CFR 50.34 and specifically deal with the physical location of the plant with respect to exclusion area boundaries and transportation routes. The TPBARs do not affect the physical location of the plant, nor are there changes to the exclusion area boundaries or transportation routes anticipated. Therefore, there is no impact on this section of the SRP.

SRP 2.1.2
Exclusion Area Authority and Control: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100.3(a), and specifically deal with the authority of the applicant to determine all activities within the designated exclusion area. The TPBARs do not impact this; the current license holder will maintain authority of the exclusion area. Therefore, this acceptance criteria will be met.

SRP 2.1.3
Population Distribution: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.34, 10 CFR 100.3, 10 CFR 100.10, and 10 CFR 100.11, and specifically deal with the population density in areas surrounding the site. The presence of TPBARs in the core is not anticipated to have a significant effect on the population density in areas surrounding the site. Therefore, a judgment is made that there is no impact.

SRP 2.2.1-2.2.2
Identification of Potential Hazards in Site Vicinity: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.34 and 10 CFR 100.10. They deal with the identification of potentially hazardous activities which are conducted at nearby industrial, military, and transportation facilities. The TPBARs do not impact these activities.

SRP 2.2.3
Evaluation of Potential Accidents: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100.10, and specifically deal with the potential for offsite hazards causing onsite accidents. The TPBARs do not impact these criteria.

SRP 2.3.1
Regional Climatology: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2 and 4 and 10 CFR 100.10(c)(2). They specifically deal with the description of the regional climate, including the ultimate heat sink meteorological data and severe regional weather phenomena. The TPBARs do not impact the regional climate.

SRP 2.3.2
Local Meteorology: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2 and 10 CFR 100.10(c)(2), and specifically deal with the local meteorological characteristics and the influence of the plant and its facilities on the local meteorological and air quality conditions. The TPBARs do not impact existing meteorological characteristics.

SRP 2.3.3
Onsite Meteorological Measurements Programs: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100.10(c)(2), 10 CFR 100.11(a), and 10 CFR 50, Appendix I, and specifically deal with the acceptability of the onsite program to measure and record meteorological data. The TPBARs do not impact the ability to measure and record meteorological data.

SRP 2.3.4
Short Term Diffusion Estimates: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100.11(a) and 10 CFR 50, Appendix A, GDC 19, and specifically deal with the atmospheric dispersion models used for calculating relative concentrations for postulated accidental releases. The evaluation includes consideration of meteorological data, site topography, release points and atmospheric diffusion parameters. The TPBARs do not impact these parameters.

SRP 2.3.5
Long-Term Diffusion Estimates: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix I, and specifically deal with the atmospheric dispersion models used for calculating relative concentrations resulting from routine releases. The evaluation includes consideration of meteorological data, site topography, release points and atmospheric diffusion parameters. The TPBARs do not impact these parameters.

SRP 2.4.1
Hydrologic Description: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100 and 10 CFR 50, Appendix A, GDC 2, and specifically deal with the completeness of the description of the hydrologic characteristics of the site including water bodies, water control structures, and water users. The TPBARs do not impact these descriptions.

SRP 2.4.2
Floods: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100 and 10 CFR 50, Appendix A, GDC 2, and specifically deal with the flood history

at the site, flood design considerations, and effects of local intense precipitation. The TPBARs do not impact these phenomena.

SRP 2.4.3
Probable Maximum Flood (PMF) on Streams and Rivers: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100 and 10 CFR 50, Appendix A, GDC 2, and specifically deal with the probable maximum flooding estimates for adjacent streams or rivers based on the probable maximum precipitation. The TPBARs do not impact these phenomena.

SRP 2.4.4
Potential Dam Failures: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100; 10 CFR 100, Appendix A; and 10 CFR 50, Appendix A, GDC 2. They specifically deal with the potential flooding resulting from postulated failures of upstream dams. The TPBARs do not impact these phenomena.

SRP 2.4.5
Probable Maximum Surge and Seiche Flooding: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100 and 10 CFR 50, Appendix A, GDC 2, and specifically deal with the identification of probable maximum hurricanes, moving squall lines or other cyclonic wind storms and the resultant flooding at the site. The TPBARs do not impact these phenomena.

SRP 2.4.6
Probable Maximum Tsunami Flooding: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100; 10 CFR 100, Appendix A; and 10 CFR 50, Appendix A, GDC 2. They specifically deal with the identification of potential tsunami generators and the resultant flooding at the site. The TPBARs do not impact these phenomena.

SRP 2.4.7
Ice Effects: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a, 10 CFR 100, and 10 CFR 50, Appendix A, GDC 2, and specifically deal with the potential for ice accumulations (i.e., ice jams, floes, etc.) and their impact on safety-related facilities and water supplies. The TPBARs do not impact these phenomena.

SRP 2.4.8
Cooling Water Canals and Reservoirs: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a, 10 CFR 100, and 10 CFR 50, Appendix A, GDC 2 and 44. They specifically deal with the design basis for capacity of the plant canals and reservoirs and their protection from various phenomena (e.g., probable maximum flooding, surges, etc.). The potential impact of the TPBARs on ultimate heat sink is discussed in the response to SRP 9.2.5 (Section 2.9.1). This must be addressed on a plant-specific basis.

SRP 2.4.9
Channel Diversions: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100 and 10 CFR 50, Appendix A, GDC 2 and 44. They specifically

deal with the potential for stream channel diversions and their impact on safety-related facilities, water supply or ultimate heat sink. The TPBARs do not impact these criteria.

SRP 2.4.10
Flooding Protection Requirements: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a, 10 CFR 100 and 10 CFR 50, Appendix A, GDC 2. They specifically deal with the definition of the flood design basis, and the identification of adequate levels of flood protection for the controlling flood conditions. The TPBARs do not impact these criteria.

SRP 2.4.11
Cooling Water Supply: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100; 10 CFR 100, Appendix A; and 10 CFR 50, Appendix A, GDC 2 and 44. They specifically deal with effects of various natural phenomena (e.g., drought, icing, etc.) on the adequacy of the cooling water supply. The TPBARs do not impact these criteria.

SRP 2.4.12
Groundwater: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55; 10 CFR 50.55a; 10 CFR 100; 10 CFR 100, Appendix A; and 10 CFR 50, Appendix A, GDC 2, 4 and 5. They specifically deal with the identification of regional and local aquifers, sources and sinks; the types and extent of onsite and offsite groundwater use; and the development of the design bases for groundwater-induced loadings on subsurface portions of safety-related structures, systems and components. The impact of the TPBARs is within normal variation.

SRP 2.4.13
Accidental Releases of Liquid Effluents in Ground and Surface Waters: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100, and specifically deal with the ability of the ground and surface water environment to delay, disperse, dilute, or concentrate accidental radioactive liquid effluent releases. Although the dispersion pathways do not change, the reference plant FSAR contains specific consideration of tritium in the released water. Although there is assumed to be a significant increase in the amount of tritium entering the primary coolant because of operation with the TPBARs, it is expected and recommended that the plant be operated with sufficient increase in primary coolant discharge such that the tritium concentrations in the primary coolant do not increase above the current normal operating levels (see Section 2.11.3). Thus, any liquid wastes that might be spilled so as to enter the ground or surface water would have tritium concentrations no different than those associated with current operation.

SRP 2.4.14
Technical Specifications and Emergency Operation Requirements: This section deals specifically with the identification of the controlling hydrologic events (e.g., dam failures, tsunamis), and any actions required to protect safety-related facilities and water supplies (e.g., sandbagging). It does not address the plant Technical Specifications or the Emergency Operation Requirements as a whole. The acceptance criteria in this section are based on the

relevant requirements of 10 CFR 50.36 and 10 CFR 50, Appendix A, GDC 2. The TPBARs do not impact the hydrologic characteristics of the site.

SRP 2.5.1
Basic Geologic and Seismic Information: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100; 10 CFR 100, Appendix A; and 10 CFR 50, Appendix A, GDC 2. They specifically deal with the documentation of the geologic, seismic, and man-made features of the region and the site. The TPBARs do not impact these criteria.

SRP 2.5.2
Vibratory Ground Motion: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100; 10 CFR 100, Appendix A; and 10 CFR 50, Appendix A, GDC 2. They specifically deal with the seismicity, the geologic and tectonic characteristics of the site and region, the correlation of earthquake activity with geologic structure or tectonic provinces, maximum earthquake potential, seismic wave transmission characteristics of the site, description of the safe shutdown earthquake, and description of the operating basis earthquake. The TPBARs do not impact these characteristics.

SRP 2.5.3
Surface Faulting: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100; 10 CFR 100, Appendix A; and 10 CFR 50, Appendix A, GDC 2. They specifically deal with the adequacy of the presented geologic and seismologic information to establish that no capable faults exist in the plant site area which would cause earthquakes to be centered there, and that there are no indications that a potential exists for surface faulting at the plant site. The TPBARs do not impact these phenomena.

SRP 2.5.4
Stability of Subsurface Materials and Foundations: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a; 10 CFR 50, Appendix A, GDC 1, 2 and 44; 10 CFR 50, Appendix B; 10 CFR 100; and 10 CFR 100, Appendix A. They specifically deal with the properties and stability of all soils and rock which may affect the plant under both static and dynamic conditions. The TPBARs do not impact these features.

SRP 2.5.5
Stability of Slopes: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a; 10 CFR 50, Appendix A, GDC 1, 2 and 44; 10 CFR 50, Appendix B; 10 CFR 100; and 10 CFR 100, Appendix A. They specifically deal with the dynamic and static stability of all slopes whose failure could adversely affect safety-related structures of the plant or pose a hazard to the public. The TPBARs do not impact these features.

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2.3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

2.3.1 INTRODUCTION

The design of the structures, components, equipment, and systems of the selected plant will, for the most part, not be impacted by the incorporation of TPBARs in the core design. The sections in Chapter 3 of the SRP deal primarily with the structural integrity of the structures, components, equipment and systems. Evaluations were performed for SRP 3.9.1, pertaining to the definition of design transients; SRP 3.9.2, pertaining to dynamic loads; SRP 3.9.3, SRP 3.9.4 and 3.9.5, pertaining to the design of the components, Control Rod Drive Mechanism and the Reactor Vessel Internals; and SRP 3.11, pertaining to the environmental qualification of equipment. Those evaluations are described in Sections 2.3.2 through 2.3.7 of this report.

SRP 3.2.1
Seismic Classification: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2 and 10 CFR 100, Appendix A. They specifically deal with the proper classification of structures, systems and components important to safety as seismic Category I. TPBARs retain the same safety classification as other burnable absorbers and thus do not impact seismic classification.

SRP 3.2.2
System Quality Group Classification: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a, and 10 CFR 50, Appendix A, GDC 1. They specifically deal with the identification of the proper quality group classifications for systems important to safety, and the requirement of quality standards commensurate with the importance of the safety function to be performed. TPBARs are a basic component of the reactor core and are therefore safety-related.

SRP 3.3.1
Wind Loadings: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, and specifically deal with the design of structures which have to withstand the effects of the design wind, and the procedures used to transform the wind velocity into an effective pressure. The TPBARs do not impact these criteria.

SRP 3.3.2
Tornado Loadings: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, and specifically deal with the design of structures which have to withstand the effects of tornadoes, the tornado parameters and the procedures used to transform those parameters into an effective loading on structures, and the postulated impact on safety-related structures or components of the failure of any structures or components not designed for tornado loads. The TPBARs do not impact these criteria.

SRP 3.4.1
Flood Protection: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, and 10 CFR 100, Appendix A, Section IV.C.

They specifically deal with the protection of structures, systems, and components important to safety from the effects flooding. The TPBARs do not impact this capability.

SRP 3.4.2
Analysis Procedures: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, and specifically deal with the adequacy of the input parameters and procedures used to transform the static and dynamic effects of the flood and groundwater into effective loads for use in the structural analysis. The TPBARs do not impact these criteria.

SRP 3.5.1.1- 3.5.1.6
Missiles: The acceptance criteria in these sections are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 3 and 4 and 10 CFR 100.10. They specifically deal with the identification of the potential for internally generated missiles, turbine missiles, externally generated missiles, and aircraft hazards, and the effect on structures, systems, and components important to safety. The TPBARs do not impact these criteria.

SRP 3.5.2
Structures, Systems, and Components to be Protected from Externally Generated Missiles: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2 and 4. They specifically deal with the identification of structures, systems, and components which need to be protected from externally generated missiles. The TPBARs do not impact these criteria.

SRP 3.5.3
Barrier Design Procedures: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2 and 4. They specifically deal with the adequacy of the design of barriers to withstand missile impact and the procedures/assumptions used in that design, specifically with respect to local damage prediction in concrete, steel, and composite sections and overall damage prediction. The TPBARs do not impact these criteria.

SRP 3.6.1
Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 4, and specifically deal with the design of structures, systems, and components important to safety to accommodate the dynamic effects of postulated pipe rupture, including the effects of pipe whipping and discharging fluids. The TPBARs do not impact these criteria.

SRP 3.6.2
Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 4. They specifically deal with the identification of postulated pipe rupture locations and configurations, and the dynamic analysis of pipe whip including determination of forcing functions of jet thrust and jet impingement. The TPBARs do not impact these criteria.

SRP 3.7.1
Seismic Design Parameters: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2 and 10 CFR 100, Appendix A. They specifically deal with the adequacy of the seismic design response spectra, the percentage of critical damping values, and the time history used in the design of seismic Category I structures, systems, and components. The TPBARs do not impact the design response spectra or the time history. The impacts of the TPBARs on the seismic response of the fuel assemblies, and the new and spent fuel storage racks are discussed in Sections 2.4.2 and 2.9.2, respectively.

SRP 3.7.2-3.7.3
Seismic System and Subsystem Analysis: The acceptance criteria in these sections are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2 and 10 CFR 100, Appendix A. They specifically deal with the adequacy of the seismic analysis methods used in the design of seismic Category I structures, systems, and components. The impacts of the TPBARs on the seismic response of the fuel assemblies and the new and spent fuel storage racks are discussed in Sections 2.4.2 and 2.9.2, respectively.

SRP 3.7.4
Seismic Instrumentation: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a; 10 CFR 50, Appendix A, GDC 2; and 10 CFR 100, Appendix A. They specifically deal with the adequacy of the seismic monitoring instrumentation, the associated inservice inspection program, and the specification of the data recording requirements. The TPBARs do not impact these parameters.

SRP 3.8.1-3.8.2
Concrete Containment/Steel Containment: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a and 10 CFR 50, Appendix A, GDC 1, 2, 4, 16 and 50. They specifically deal with the description of the containment; the applicable codes, standards, and specifications; loads and loading combinations; design and analysis procedures; structural acceptance criteria; materials, quality control, and special construction techniques; and testing and inservice surveillance requirements. The TPBARs do not impact these parameters.

SRP 3.8.3
Concrete and Steel Internal Structures of Steel or Concrete Containments: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a and 10 CFR 50, Appendix A, GDC 1, 2, 4, 5 and 50. They specifically deal with the description of the internal structures; the applicable codes, standards, and specifications; loads and load combinations; design and analysis procedures; structural acceptance criteria; materials, quality control, and special construction techniques; and testing and inservice surveillance requirements. The TPBARs do not impact these structures.

SRP 3.8.4
Other Seismic Category I Structures: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a; 10 CFR 50, Appendix A, GDC 1, 2, 4 and 5; and 10 CFR 50, Appendix B. They specifically deal with the description of the structures; the

applicable codes, standards, and specifications; loads and loading combinations; design and analysis procedures; structural acceptance criteria; materials, quality control, and special construction techniques; testing and inservice surveillance requirements; and masonry walls. The TPBARs do not impact these structures.

SRP 3.8.5
Foundations: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a and 10 CFR 50, Appendix A, GDC 1, 2, 4 and 5. They specifically deal with the description of the foundation; the applicable codes, standards, and specifications; loads and load combinations; design and analysis procedures; structural acceptance criteria; materials, quality control, and special construction techniques; and testing and inservice surveillance requirements. The TPBARs do not impact these parameters.

SRP 3.9.1
Special Topics for Mechanical Components: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 1, 2, 14 and 15; 10 CFR 50, Appendix B; and 10 CFR 100, Appendix A. They specifically deal with the development of a list of transients to be used in design and fatigue analysis and the identification of the stress analysis methods to be used. An investigation of the potential impact of the TPBARs on the design transients was performed and is described in Section 2.3.2 of this document. There are no changes in the stress analysis methods used.

SRP 3.9.2
Dynamic Testing and Analysis of Systems, Components, and Equipment: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 1, 2, 4, 14 and 15. They specifically deal with the testing and analyses employed to assure the structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports under vibratory loadings, including those due to fluid flow and postulated seismic events, and a combination LOCA and SSE. An evaluation of the impact of the TPBARs on component response to dynamic loads is summarized in Section 2.3.3 of this document.

SRP 3.9.3
ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a and 10 CFR 50, Appendix A, GDC 1, 2, 4, 14 and 15. They specifically deal with the design and service loading combinations for Code Class 1, 2, and 3 items, design and installation of pressure relief devices, and the design and structural integrity of Code Class 1, 2, and 3 component supports. Evaluations of the impact of the TPBARs on the structural integrity of the reactor coolant pumps, reactor coolant system piping and supports, pressurizer, steam generators and reactor vessel are summarized in Section 2.3.4 of this document.

SRP 3.9.4
Control Rod Drive Systems: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a and 10 CFR 50, Appendix A, GDC 1, 2, 14, 26, 27 and 29. They specifically deal with the adequacy of the CRDS description provided, the

identification of applicable codes and standards, the appropriate combinations of design loads, the design stress limits and allowable deformations, and the adequacy of the operability assurance program. An evaluation of the impact of the TPBARs on the CRDS is summarized in Section 2.3.5 of this document.

SRP 3.9.5
Reactor Pressure Vessel Internals: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a and 10 CFR 50, Appendix A, GDC 1, 2, 4 and 10. They specifically deal with the description of the configuration and general arrangement of the mechanical and structural internal elements; the design procedures and criteria used, including design and service stress limits; the allowable deformation limits; and the acceptability of calculated stresses and deformation with respect to the limits. An evaluation of the impact of the TPBARs on the reactor internals is summarized in Section 2.3.6 of this document.

SRP 3.9.6
Inservice Testing of Pumps and Valves: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a and 10 CFR 50, Appendix A, GDC 37, 40, 43, 46 and 54. They specifically deal with the acceptability of the pump and valve test programs, including compliance with appropriate sections of the ASME Code. The inservice test programs for pumps and valves are not impacted by the presence of TPBARs.

SRP 3.10
Seismic and Dynamic Qualification of Mechanical and Electrical Equipment: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100, Appendix A; 10 CFR 50, Appendix A, GDC 1, 2, 4, 14 and 30 and 10 CFR 50 Appendix B. They specifically deal with the acceptability of the equipment seismic qualification program. The TPBARs do not impact this program.

SRP 3.11
Environmental Qualification of Mechanical and Electrical Equipment: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 4 and 10 CFR 50, Appendix B. They specifically deal with the acceptability of the required equipment performance under normal, abnormal, accident, and postaccident environment conditions, and the adequacy of the qualification program employed for this. An evaluation of the impact of TPBARs on the equipment environment is summarized in Section 2.3.7 of this document.

2.3.2 NSSS DESIGN TRANSIENTS (SRP 3.9.1) EVALUATION

2.3.2.1 Introduction

The NSSS design transients are used as an input for the component fatigue stress analyses of the various RCS components (reactor vessel and internals, steam generators, RCS piping, reactor coolant pumps, and pressurizer). They describe the thermal and hydraulic (i.e., pressure, temperature, and flow) variations that occur during various normally expected plant

maneuvers and during unanticipated transients. The expected frequency of occurrence for each design transient is developed and supplied for use in the component fatigue analyses.

The design transients can be affected by changes in the rated power level, primary side operating temperature (i.e., programmed T_{AVG} value), primary side operating pressure (i.e., pressurizer pressure setpoint), steam generator tube plugging level, RCS major component replacement (e.g., steam generator replacement), RCS flow changes (potentially a consequence of steam generator tube plugging, reactor coolant pump impeller shaving, reactor vessel internals or fuel changes, or steam generator replacement), fuel reactivity changes, or changes to plant procedures resulting in a new type of design transient which was not part of the original design transient package (e.g., removing feedwater heaters from service, T_{AVG} coastdown, or load regulation).

The TPC could impact the following areas, which would then require a modification of the design transients that are applicable to the reference plant:

- RCS operating temperature
- RCS operating pressure
- RCS flow
- Fuel reactivities
- Plant procedures

Therefore, these areas were reviewed to determine whether they are impacted by the TPC design.

2.3.2.2 Methodology

As described in the introduction section, there are several parameter changes that could require modification of the RCS design transients. The methodology used was to compare the parameters identified for the TPC with the existing parameters for the reference plant. If the parameters do not change significantly, then it will be concluded that the TPC will not impact the design transients, and the existing design transients for the reference plant are applicable for the TPC. If the parameters for the TPC are significantly changed with respect to the currently licensed reference plant parameters, then the design transients will be modified.

2.3.2.3 Significant Input Parameters and Assumptions

The following were the key input parameters and assumptions used for this evaluation.

1. NSSS performance parameters for the TPC plant in Table 1-1. This table describes the rated power level, RCS operating temperature and pressure, RCS flow, steam generator type, and steam generator secondary side steam pressure, steam/feedwater flow, and feedwater temperature.

2. NSSS performance parameters for the reference plant (see Table 1-1).

3. Fuel reactivity parameters for the TPC.

4. Based upon the following assumptions, the design basis operational behavior of the plant was not revised as a result of incorporation of the TPBARs.

   a. The TPBAR, inserted into the thimble, is designed to meet the requirements of a large representative four-loop Westinghouse reactor under Conditions I, II, III, and IV, defined in the Safety Analysis Report of the reference plant.

   b. The incorporation of the TPBARs will neither cause nor create any new accident scenarios other than those currently identified in Regulatory Guide 1.70 and the NRC Standard Review Plan or add to the probability of existing scenarios.

   c. There will be no plant power uprating (or derating) or RCS thermal-hydraulic parameter changes made to enhance reactor performance or operational flexibility for the TPC.

2.3.2.4 Results

A comparison of the design operating parameters for the TPC plant vs. the parameters for the reference plant indicated no differences in the following parameters.

- Rated power level
- RCS operating temperature
- RCS operating pressure
- RCS flow
- Type of steam generator
- Steam generator secondary side pressure
- Steam/feedwater flow
- Feedwater temperature

In addition, a review of the fuel reactivities indicated that they are representative of those to be expected in a standard Westinghouse 4-loop plant.

Based on the fact that there are no significant differences in the parameters discussed above for the TPC plant versus the reference plant, there is no need to revise the RCS design transients.

2.3.2.5 Potential Plant Specific Analysis

For a specific plant implementation, an evaluation must be performed to verify that none of the parameters (as listed in Section 2.3.2.4) are impacted. Significant changes require a revision of the NSSS design transients for the implementing plant and change the input to the NSSS component structural analyses or evaluations. If no changes to the NSSS design transients are necessary, then a statement of "no impact" will be issued to the structural analysts.

2.3.3 DYNAMIC LOADS (SRP 3.9.2) EVALUATION

2.3.3.1 Large Break LOCA Forces

The effect of TPBARs on hydraulic forces during a large break LOCA has been evaluated on both a best estimate basis and on a design parameter basis. The incorporation of TPBARs affects LOCA forces because they slightly increase reactor vessel hydraulic resistance and thus reduce the best estimate primary loop flow rate. This results in a small increase in core temperature rise with an associated increase in hot leg temperature and decrease in cold leg temperature. The most significant of these effects is the decrease in cold leg temperature which increases the subcooled break flow rates and the limiting hydraulic forces associated with the cold leg break.

In this evaluation, it was assumed that all assemblies not under control rods receive TPBARs so that the overall hydraulic resistance increase is similar to that produced by the thimble plugs originally installed in all unused fuel assembly thimble locations. Installation of thimble plugs in this fashion results in a 0.7% decrease in loop flow rate, compared to the case with no thimble plugs or TPBARs. For a typical 3 or 4 loop plant, this results in a decrease in cold leg temperature of 0.20 to 0.25°F and an increase in the peak horizontal LOCA forces of less than 0.2%.

From the point of view of design basis parameters, such as Thermal Design Flow and Reactor Coolant Temperatures, sufficient margin is often available to cover a change of the magnitude indicated above. Thus, for a typical plant the incorporation of TPBARs is not expected to result in changes in these parameters. In the case where margin is not available to accommodate the change and the design parameters are affected, the increase in peak horizontal LOCA forces of 0.2% is of a small enough magnitude that margin in break size estimates is often available to preclude the need to review the ASME code design analysis reports. It is noted that fuel assembly thimble plugs were originally installed in all plants and were included in the original design LOCA forces analysis. Thus, installation of TPBARs moves the plant conditions toward the original design analysis basis.

As a result of the above considerations, on a generic basis, it is unlikely that the incorporation of TPBARs would require any re-analysis of the LOCA forces and the associated rework of the ASME code stress analyses. The LOCA forces for the reference core were calculated for a primary system temperature window with a lower bound cold leg temperature well below the value listed in Table 1-1 for the TPC. Thus, the LOCA forces calculated for the reference core continue to be bounding for the TPC.

For specific plant implementation of the TPC, a confirming evaluation based on selected RCS parameters is recommended. The results of this evaluation will provide input to the reactor vessel, reactor internals, and reactor coolant piping/supports structural analyses.

2.3.3.2 Flow Induced Vibration

With regard to Flow Induced Vibrations, the response of a given set of reactor vessel internals is generally influenced by: a) hydraulic design parameters such as flow rates and vessel/core inlet

and outlet temperatures, and b) changes in the dynamic characteristics of the fuel assemblies. The mechanical design flow rates and the coolant inlet and outlet temperatures remain unchanged from the reference plant configuration. As noted in Section 2.4.2.1.4, the TPBAR has an insignificant effect on the dynamic characteristics of the fuel assembly. Consequently, the structural integrity of the reactor vessel internals with the TPC will not be adversely impacted with regard to Flow Induced Vibrations. In addition, the vibration characteristics of the TPBARs within the guide thimbles are addressed in Section 3.4.6.

2.3.4 COMPONENTS (SRP 3.9.3) EVALUATIONS

2.3.4.1 Steam Generator Components

This section evaluates the potential impact of the TPC on the structural analysis and thermal-hydraulic characteristics of the steam generators in the reference plant. The operating conditions and design transients were reviewed to identify any changes caused by the TPC and determine their impact on the structural analysis and thermal-hydraulic characteristics of the steam generators.

The NSSS performance parameters for the proposed TPC plant, provided in Table 1-1, are the same as those currently licensed for the reference plant. Per Section 2.3.2, the plant design transients will not change due to the TPC and neither will their frequency of occurrence. Therefore, the structural and the thermal hydraulic analyses performed for the steam generator components of the reference plant remain applicable, and no new analyses are necessary. The steam generator components satisfy the requirements of the applicable ASME Code for the licensed conditions for the reference plant, and will continue to do so for the TPC plant. Current steam generator documentation will remain applicable.

The conclusions of the present evaluation apply to all PWR steam generators of Westinghouse design as long as the design operating conditions and NSSS design transients are not affected by the TPC design. For specific plant implementation, the impact for the RCS parameters and NSSS design transients will need to be reviewed to determine the impact for the steam generator structural evaluation. No significant impact is expected for TPC implementation.

2.3.4.2 Pressurizer Components

The pressurizer vessel structural analysis for the reference plant was evaluated to determine if the key inputs change as a result of the TPC.

The limiting operating conditions of the pressurizer occur when the RCS pressure is high and the RCS hot leg (T_{HOT}) and cold leg (T_{COLD}) temperatures are low. This maximizes the temperature differential, dT, between pressurizer and hot or cold leg fluid that is experienced by the pressurizer. Due to flow in and out of the pressurizer during various transients, the surge nozzle alternately sees water at the pressurizer temperature (T_{PZR}) and water from the RCS hot leg at T_{HOT}. If the RCS pressure is high (which means, correspondingly, that T_{PZR} is high) and T_{HOT} is low, then the surge nozzle will see maximum thermal gradients; and, thus experience the maximum thermal stress. Likewise, the spray nozzle and upper shell temperatures alternate between steam at T_{PZR} and spray, which, for many transients, is at T_{COLD}.

Thus, if RCS pressure is high (T_{PZR} is high) and T_{COLD} is low, then the spray nozzle and upper shell will also experience the maximum thermal gradients and thermal stresses.

The pressurizer structural evaluation was performed by comparing the key inputs in the current pressurizer stress report for the reference plant with the corresponding key inputs for the TPC. The following key inputs were included in the evaluation.

- T_{HOT}, T_{COLD}, and T_{PZR} (Table 1-1)
- Design Transients (Section 2.3.2)

T_{HOT}, T_{COLD}, and T_{PZR} were found to be identical for the reference plant and the TPC plant. Furthermore, the evaluation in Section 2.3.2 shows that the transients for the reference plant pressurizer are not impacted by the TPC.

Based upon these observations, it is concluded that the reference plant pressurizer stress analysis envelopes the TPC plant parameters. Therefore, no additional stress/fatigue/fracture mechanics analyses are required for the pressurizer components for the TPC.

For specific plant implementation, the RCS parameters and NSSS design transients need to be reviewed to determine any impact for the pressurizer. No significant impact is expected for TPC implementation.

2.3.4.3 RCS Piping and Supports

The impact of the TPC installation on the RCS piping and supports is assessed by evaluating the changes to the NSSS design transients with respect to temperatures, pressures and frequency of occurrence.

Acceptance criteria for the RCS piping and supports include limiting stresses to ASME Section III allowables, and limiting fatigue usage factors less than 1.00.

The following factors are used as input to this assessment:

- RCS design parameters (power, flows, temperatures, and pressures) do not change to accommodate the TPC because the existing representative plant parameters bound operation with TPC. The representative plant parameters, though not bounding for all PWRs, were conservatively selected to be bounding for candidate plants with a high degree of confidence.

- The TPBARs have a negligible impact on the Large and Small Break loss of coolant accident (LOCA) physical response characteristics (Section 2.15), the LOCA forces (Section 2.3.3), and Steam Generator Tube Rupture event. (Section 2.15)

- An evaluation performed to determine the impact of the TPC on the design transients for the representative plant concluded that there are no changes to the RCS parameters important to the design transients, nor are there significant changes to the core reactivities,

relative to the representative plant typical values. Therefore, the representative plant design transients continue to apply (Section 2.3.2).

The result of this assessment is that there is no impact due to the TPC on the RCS piping and support stress and fatigue analyses, or on leak-before-break evaluations, since the existing NSSS design transients and RCS parameters continue to be applicable with TPC installation. There are no affected documents (e.g., piping stress reports) as a result of TPC installation.

This assessment is expected to be generally applicable to candidate PWRs, since the representative plant has a high power rating and therefore conservative plant parameters.

For specific plant implementation, statements of impact for the LOCA evaluations, RCS parameters, and NSSS design transients need to be reviewed in order to determine the impact for the RCS piping and supports. No significant impact is expected for TPC implementation.

2.3.4.4 Reactor Coolant Pumps

The RCP structural analysis considers the operating temperatures and pressures of the coolant including transient conditions, and other sources of loading on the pressure boundary by way of seismic and LOCA conditions. The evaluation of the effect of using TPBARs was performed by examining the changes in the key input parameters for the structural analysis, and then determining the impact on the analysis.

RCPs are subject to cold leg transients. Thus, the normal operating temperature is defined as the steam generator outlet temperature in Table 1-1. The design transients do not change per Section 2.3.2. Typical LOCA forces would not increase by more than 0.2% per Section 2.3.3.1. Seismic forces are dependent upon site location only.

The acceptance criteria are given in the ASME Boiler and Pressure Vessel Code, Section III for Class 1 components. Compliance with these criteria is demonstrated by existing analyses.

The evaluation determined that the change in the reference plant operating temperature is bounded by existing analyses. The design transients do not change, thus the current analysis continues to apply.

The RCP hydraulics and motor function is acceptable for the TPC. The best estimate loop flow rate decrease associated with the TPBARs is small and is bounded by the original plant parameters. The slight decrease in the best estimate pump operating temperature has minimal and acceptable impact on its performance.

In conclusion, the RCPs for the reference plant are not affected by the TPC.

For specific plant implementation, statements of impact for the LOCA evaluations, RCS parameters, and NSSS design transients need to be reviewed in order to determine the impact for the RCPs. No significant impact is expected for TPC implementation.

2.3.4.5 Auxiliary Heat Exchangers, Tanks, Pumps and Valves

The purpose of this evaluation is to determine the effect of the TPC on the design/operation of auxiliary equipment consisting of heat exchangers, tanks, pumps and valves supplied as part of the NSSS. The design requirements included steady state conditions as well as transient conditions, where applicable.

All of the safety-related equipment and systems ensure one or more of the following functions:

- the integrity of the reactor coolant pressure boundary,
- the capability to shut down the reactor and maintain it in a safe shutdown condition, or
- the capability to prevent or mitigate the consequences of accidents that could result in potential offsite limits comparable to the 10 CFR Part 100 guidelines.

The applicable auxiliary equipment design transients for the reference plant were reviewed. The only transients that could be potentially impacted by the TPC are those temperature transients that are impacted by the full load NSSS operating temperatures, namely T_{HOT} and T_{COLD}. These transients are based on an assumed full load NSSS T_{HOT} and T_{COLD} of 650°F and 560°F, respectively. These original design NSSS temperatures were selected to ensure that the resulting design transients would be conservative for a wide range of NSSS operating temperatures.

A comparison of the NSSS operating temperatures shows that the proposed operating temperatures for the TPC (T_{HOT} and T_{COLD} of 620°F and 556.8°F, respectively, from Table 1-1) remain bounded by those used to develop the design transients. Consequently, the actual temperature transients (i.e., the change in temperature from T_{COLD} dictated by the reference plant parameters to a lower auxiliary system related temperature, or vice versa) are less severe than the design temperature transients. Therefore, the current design transients are still bounding for the TPC at the NSSS operating conditions specified in Table 1-1.

Based on the above and the design transient information in Section 2.3.2, no input parameters or assumptions have changed from the analysis of record as a result of the TPC. Therefore all of the analyses of record still remain valid. However, the equipment in the spent fuel pit system required additional evaluation, due to the potential increase in the spent fuel pit heat load (see Section 2.9.3). It was determined that all spent fuel pit components for the reference plant have a design temperature of 190 to 200°F, and therefore, the revised temperature is bounded by the component design temperatures.

Based upon the fact that the heat exchangers, tanks, pumps and valves are not affected by the TPC, they are still acceptable based upon their original design requirements and operability constraints. Therefore, there are no new limitations associated with auxiliary heat exchangers, auxiliary tanks, pumps, and valves due to the implementation of the TPBARs. No documentation with respect to any of the components needs to be changed as part of this effort.

For specific plant implementation, a review should be performed for all spent fuel pit components based on plant specific parameters. In addition, a confirming check should be performed to verify that the implementing plant's operating temperatures for the TPC remain within the conservatively established temperature bounds of the auxiliary systems design transients.

2.3.4.6 Reactor Vessel Structural Evaluation

The reactor vessel structural evaluation was performed by comparing the key inputs in the current reactor vessel stress report for the reference plant with the corresponding key inputs for the TPC.

The following key inputs were included in the evaluation:

- reactor vessel normal operating temperatures
- NSSS design transients
- reactor vessel/reactor internals interface loads
- gamma heating rates at the vessel shell

Information concerning how each of the key inputs would vary with the implementation of the TPC was evaluated. The evaluations of the information on the individual inputs are summarized in the following paragraphs.

The reference plant reactor vessel is analyzed for an operating plant temperature differential that envelopes the vessel operating temperature differential for the TPBAR program. No additional reactor vessel thermal and structural analyses are warranted by the operating temperature changes.

The NSSS design transients are unchanged as a result of the full core TPBAR implementation (Section 2.3.2). Therefore, the design transients do not necessitate a revision to the current reactor vessel stress report for the reference plant.

The reactor vessel/reactor internals interface seismic and LOCA loads for the TPC are enveloped by the corresponding design interface loads which are already considered in the reactor vessel stress report for the reference plant (see Section 2.3.6). Therefore, no additional structural analysis to resolve the loading at the reactor vessel/reactor internals interfaces (main closure/core barrel and upper support plate flanges; outlet nozzles/core barrel nozzles; core support lugs/lower radial support keys) is necessary.

Finally, the evaluation in Section 2.3.6.6 concluded that the gamma heating rates in the baffle/barrel region resulting from the TPC are bounded by the gamma heating effects previously considered in the region. It is, therefore, concluded that the gamma heating at reactor vessel shell outboard of the baffle/barrel region is also within a range that was previously considered. Therefore, a detailed evaluation of the reactor vessel core region shell for the structural effects of gamma heating resulting from the TPC is not necessary.

Based upon the evaluation results, no revisions to the reactor vessel stress report for the reference plant are necessary for TPC implementation.

There are no reactor vessel structural analysis issues with regard to full core TPBAR implementation for the reference plant. However, there could be plant-specific issues for earlier vintage plants with lesser design bases. For example, reactor vessel/reactor internals interface loads were not completely identified for some earlier plants. For specific plant implementation, statements of impact for the RCS temperatures, NSSS design transients, reactor vessel/internals interface loads, and gamma heating rates at the vessel shell need to be reviewed. This is to determine whether there is any impact for the reactor vessel structural evaluation. No significant impact is expected for TPC implementation.

2.3.5 CONTROL ROD DRIVE MECHANISM (SRP 3.9.4) EVALUATION

The Control Rod Drive Mechanism (CRDM) structural analysis considers the operating temperatures and pressures of the coolant including transient conditions, and other sources of loading on the pressure boundary by way of seismic and LOCA conditions. The reference core and the TPC have 53 full-length control rod assemblies. The evaluation of the effect of the TPC was performed by examining the changes in the key input parameters for the structural analysis, and then determining the impact on the analysis.

The CRDMs are subject to cold leg transients. Thus, the normal operating temperature is defined as the Vessel Inlet temperature in Table 1-1. The reference plant design transients apply without change per Section 2.3.2.4. Typical LOCA forces would not increase by more than 0.2% per Section 2.3.3.1. Seismic forces are dependent upon site location only.

The acceptance criteria are given in the ASME Boiler and Pressure Vessel Code, Section III for Class 1 components. Compliance with these criteria will be demonstrated by existing analyses supplemented by additional calculations or evaluations as required.

The vessel inlet temperature from Table 1-1 is 556.8°F. The analysis of the reference plant CRDMs shows that there is ample margin for the vessel inlet temperature (approximately 18° F remains). Thus the vessel inlet temperature for the TPC, which is unchanged with respect to the current design temperature, remains bounded by the limit.

Section 2.3.2.4 concludes that the reference plant design transients do not change, thus the generic CRDM analysis continues to apply.

LOCA and seismic loading conditions are resolved in concert with the Reactor Pressure Vessel System Analysis (RPVSA). RPVSA applied site-specific LOCA and seismic loads to a dynamic model for the reference plant and determined the loading on the CRDM. The applied CRDM loads were then compared to the allowable generic loads and found to be acceptable. No further analysis is required.

The CRDMs satisfy the ASME Boiler and Pressure Vessel Code requirements with justification provided by the generic stress report, the analysis of record, and the LOCA and seismic load evaluations.

For specific plant implementation, statements of impact for the RCS parameters and NSSS design transients should be reviewed to determine whether there is any impact for the CRDM structural analysis. No significant impact is expected for TPC implementation.

2.3.6 REACTOR INTERNALS (SRP 3.9.5) EVALUATION

2.3.6.1 Introduction

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

Figure 2.3.6-1 illustrates the various components and features of the reactor internals system for the reference plant. The lower reactor internals assembly supports the fuel assemblies on the sides and at the bottom and consists of the lower core plate, lower support plate, secondary core support structure and the baffle/barrel region components. The guidance and alignment of the lower core support assembly during insertion into the reactor vessel is provided by the radial support system, the head-vessel alignment pins and special temporary guide studs attached to the vessel. The holddown spring rests on top of the flange of the lower core support assembly. The upper core support assembly consists of the upper support plate, upper support columns, and upper core plate, and rests on top of the hold down spring. The guidance and alignment of the upper core support assembly during its insertion is provided by the head-vessel alignment pins, the upper core plate alignment pins in the core barrel assembly and the special temporary guide studs attached to the vessel. The alignment of the core, i.e., each fuel assembly, is provided through the engagement of the fuel pins with the lower support plate and the upper core plate. The vessel upper head compresses the holddown spring providing joint preload.

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

2.3.6.2 Thermal Hydraulic Methodology

The thermal hydraulic analytical studies predict the reactor vessel pressure losses by classical analytical fluid mechanics. The following continuity and momentum equations are solved for a flow system that represents the entire reactor vessel and internals system:

$$W = \rho VA = \text{constant}$$

$$P_j = P_i + \sum_i^j (K + fl/D) \frac{\rho V^2}{2 g_c}$$

where:

W	=	mass flow rate,
f	=	fluid density,
V	=	fluid velocity,
A	=	area,
P	=	pressure,
K	=	loss factor,
f	=	friction factor,
l	=	length,
D	=	diameter, and
g_c	=	gravitational constant.

The reactor pressure vessel system evaluation was performed based on a full core of Westinghouse 17x17 VANTAGE+ fuel with IFM grids.

2.3.6.3 Thermal Hydraulic Results

Pressure Drop

This part of the evaluation assesses the impact of the TPC on the pressure losses experienced by the primary coolant as it flows through the reactor vessel. This is accomplished by comparing the pressure losses for the reference plant with those for the TPC plant. The pressure drops for the reference plant with and without TPBARs at mechanical design flow rates are summarized in Table 2.3.6-1. The addition of TPBARs increases the core loss coefficient to [ ][b] from [ ][b] without TPBARs. This increase in the core loss coefficient causes the core pressure drop to increase from [ ][b] to [ ][b] psi across the core. Also, the upper core plate loss coefficient increases due to the TPBARs, and this causes the pressure drop across the upper core plate to increase. The comparison in Table 2.3.6-1 shows the impact of the TPBARs on the pressure

drops. The pressure drop data are normally input to the LOCA and non-LOCA safety analyses and to overall NSSS performance calculations to assure that design parameters remain bounding. The effect of the TPBARs was sufficiently small that the design parameters remain bounding.

Upper Head Fluid Temperature

The temperature of the upper head fluid volume is one of the critical parameters used in the analyses performed for the Loss of Coolant accident (LOCA). The reference plant has been designed to maintain the temperature of the primary coolant in the upper head at the inlet reactor coolant temperature (T_{COLD}). The analysis models the interaction between all different flow paths into and out of the upper head region and therefore can be used to determine the amount of flow required to ensure the upper head volume will be at T_{COLD} during normal reactor operation.

Calculations determined that the spray nozzle flow into the upper head during normal reactor operation is greater than the flow required to ensure that the upper head volume will be at T_{COLD}. The results of the evaluation showed that the predicted bulk closure head fluid temperature of the TPC during normal reactor operation will not differ from the T_{COLD} value.

Hydraulic Lift Forces

An evaluation of the hydraulic lift forces on the reactor internals was performed to determine the impact of the TPC on the hydraulic lift forces. To ensure that the internals assembly remain seated and stable on the reactor vessel ledge during normal reactor operation the reactor internals hydraulic lift forces combined with other mechanical and body forces must not exceed the preload of the core barrel flange against the reactor vessel.

The tabulation of the reactor internals hydraulic lift forces for the reference plant and the TPC plant are presented in Table 2.3.6-2. The hydraulic lift forces were calculated at the Mechanical Design rate of 416,800 gpm total vessel flow. Table 2.3.6-1 shows a higher Δp across the upper core plate for the TPC than for the reference plant. This is reflected in the higher hydraulic lift force on the upper core plate (see Table 2.3.6-2). However, the total force on the lower package and upper package is less for the TPC; [ ][b] lbs versus [ ][b] lbs for the lower package and [ ][b] lbs versus [ ][b] lbs for the upper package. With the TPC, the reactor internals hydraulic lift forces that impact the core barrel preload are less than the case with the reference core which includes the net force on the core barrel flange, preventing liftoff.

Pressure Relief Hole Velocity

The baffle/barrel region design for the reference plant is the standard Westinghouse upflow design. This design includes four (4) levels of holes in the baffle plates known as baffle plate pressure relief holes. These holes were included to enhance the depressurization of the baffle/barrel region during a Loss of Coolant Accident (LOCA). The maximum velocity between the fuel rods opposite the pressure relief holes must be less than the [ ][b] ft/sec criterion for the reference fuel assembly design. These holes are modeled in the calculation and velocities through these holes and between the fuel rods are calculated.

Calculations were performed with TPBARs and the maximum velocity between the fuel rods opposite these pressure relief holes was determined to be [ ]^{b} ft/sec based on nominal hydraulic characteristics of the fuel/internals system at 100% power steady state conditions. Therefore the maximum velocity criterion of less than [ ]^{c} ft/sec has been met.

Core Bypass Flow Confirmation

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

For the reference plant, the present design core bypass flow criteria is 8.4% of the total vessel flow. Calculations were performed to ensure that the design value of 8.4% can be maintained.

The principal core bypass flow paths are:

1. Baffle/Barrel Region

   The reference plant reactor vessel internals configuration incorporates a coolant upflow in the region between the core barrel and the baffle plates. In this configuration, a majority of the coolant exits the reactor vessel inlet nozzles and flows downward in the annulus between the vessel and core barrel. The downward flow passes over the thermal neutron pads to the lower plenum, turns and flows up through the core region. At the entrance to the core region, a portion of the coolant flow is diverted to enter and cool the core barrel-baffle region. This diverted flow passes through machined holes in all the former plates. There is flow communication between the core region and the baffle/barrel region flow through the baffle plate to baffle plate gaps and through the baffle plate pressure relief holes. At the top of the baffle/barrel region, the flow combines with the main coolant stream flowing up through the core region. Therefore, in this upflow design, all the flow that passes through the baffle barrel region is core bypass flow since this flow is considered to never pass through the core.

2. Vessel Head Cooling Spray Nozzles

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

3. Core Barrel - Reactor Vessel Outlet Nozzle Gap

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

4. Fuel Assembly - Baffle Plate Cavity Gap

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

5. Fuel Assembly Thimble Tubes

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

Fuel assembly hydraulic characteristics and system parameters, such as inlet temperature, reactor coolant pressure and flow, were used to determine the bypass flows. Calculations were performed to assess the core bypass flows for nominal conditions. The results of the core bypass flow evaluation for the reference plant show that the calculated TPC core bypass flow, including uncertainties, is [ ]^b % of the total vessel flow. Therefore, the design core bypass flow value of 8.4% of the total vessel flow can be maintained, and will be used as input for mechanical and safety evaluations.

2.3.6.4 Thermal Hydraulic Conclusions

Summarized below are the results of the reactor pressure vessel system thermal hydraulic evaluation.

Table 2.3.6-1 shows the impact of the TPBARs on the pressure drops. The pressure drop data is necessary input to the LOCA and non-LOCA safety analyses and to overall NSSS performance calculations.

The predicted bulk closure head fluid temperature during normal reactor operation with the TPC will not differ from the T_{COLD} value.

With the TPC the reactor internals hydraulic lift forces that impact the core barrel preload are less than the reference core case.

The maximum velocity between the fuel rods opposite the pressure relief holes was determined to be less than the [ ]^{b} ft/sec criterion.

The design core bypass flow value of 8.4% of the total vessel flow can be maintained, and is the basis for the mechanical and safety evaluations.

The analysis confirms that the TPC has a minor impact on the thermal/hydraulic results. Overall, the addition of the TPBARs has the same thermal hydraulic impact as having thimble plugs in-place.

Table 2.3.6-1
Reactor Vessel/Internals/Fuel Pressure Drop for TPC Plant and Reference Plant
Reactor Pressure Vessel Pressure Drops (psi) @ Mechanical Design Flow

Region	TPC Plant	Reference Plant

b

Mechanical Design Flow = 416,800 gpm (total vessel flow)

Table 2.3.6-2
Maximum Hydraulic Lift Forces for TPC Plant and Reference Plant
Maximum Hydraulic Lift Forces (lbs.) @ Mechanical Design Flow

Mechanical Design Flow = 416,800 gpm (total vessel flow)

Region	TPC Plant	Reference Plant

b

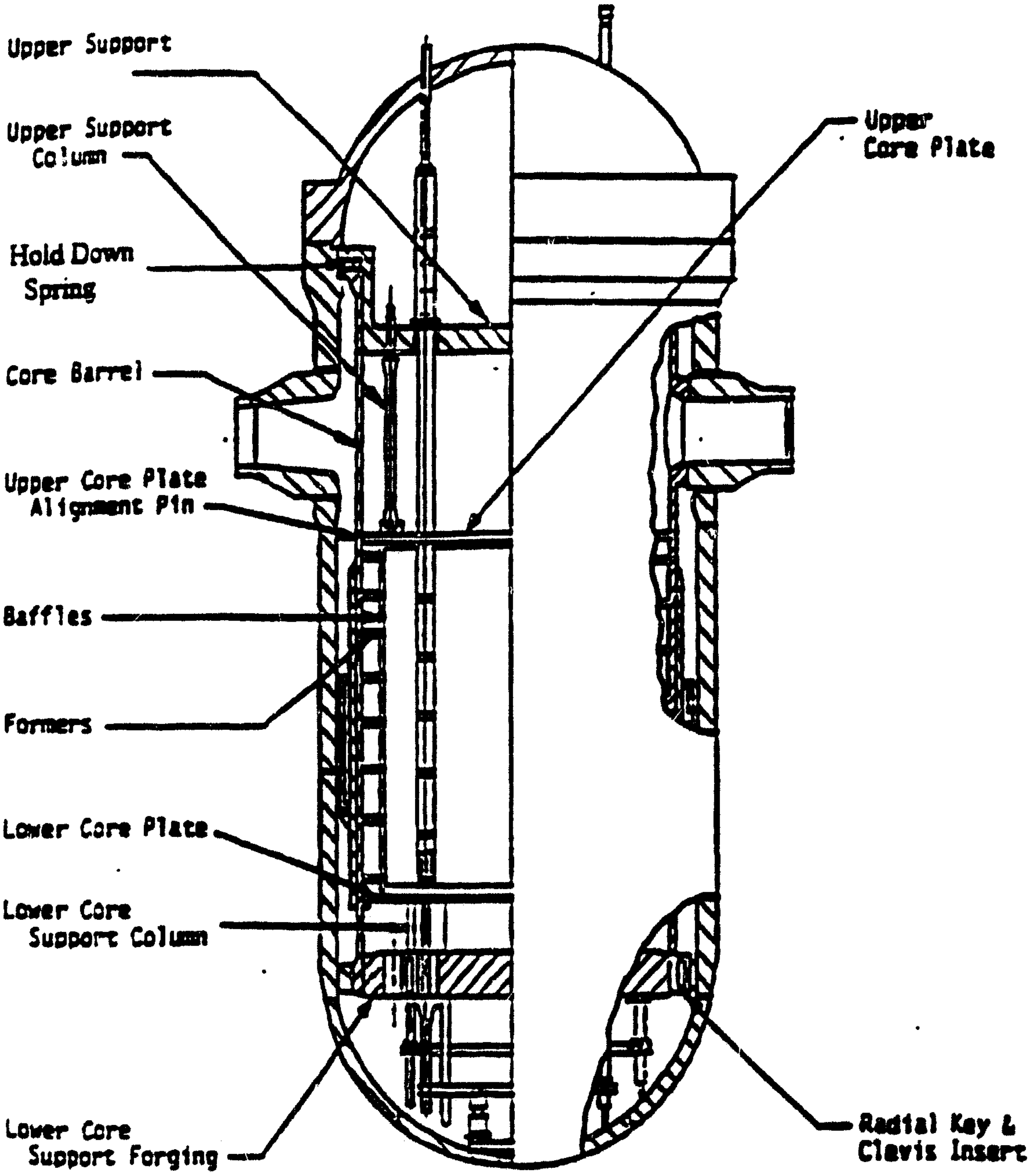

Figure 2.3.6-1
Reference Plant Reactor Internal Components

2.3.6.5 Reactor Internals Structural Integrity Methodology

The components of the reactor vessel internals are exposed to the environmental conditions of the reactor coolant system. Evaluations related to the structural integrity of the reactor internals consider all relevant environmental conditions. The reactor vessel internals are in contact with the primary coolant, consequently the integrity of these components is affected by the pressure and temperature conditions of the primary coolant. Core support structures such as the baffle, former and core support plates which are located closest to the nuclear fuel are also exposed to high levels of radiation during operation.

In general, changes in the nuclear fuel design require a reevaluation of the reactor vessel internals. An evaluation was performed to determine the impact that the TPC will have on the structural integrity of the reactor core support structures.

The methodology of the evaluation was to determine the impact of the TPC on parameters which have the greatest impact on the structural integrity of the reactor internals. For the evaluation performed, conditions which affect the reactor internals are divided into four categories:

- Fuel assembly dynamic characteristics
- RCS thermal hydraulics
- Gamma heating
- LOCA and seismic

Any changes in these conditions resulting from the TPC are assessed. Determining that the TPC conditions are enveloped by previously assumed conditions or, that resulting differences have a negligible impact on the structural integrity of the reactor internals, demonstrates that the reference plant's existing reactor internals structural evaluations are applicable and bounding for the TPC.

2.3.6.6 Reactor Internals Structural Integrity Evaluation

Fuel Assembly Dynamic Characteristics

The evaluation described in Section 2.4.2.1.5 concluded that the TPBAR has no impact on fuel assembly structural integrity and that the fuel assembly response to LOCA and seismic events will not be affected for the reference plant incorporating a TPC. As a result, the upper and lower core plate interface loads during LOCA and seismic events will also not be affected by the TPC.

RCS Thermal Hydraulics

The analysis described in Section 2.3.6.2 - 2.3.6.4 shows that the implementation of TPBARs in the reference plant has only minor effects on the reactor vessel thermal hydraulics. The overall effect is similar to the previously evaluated condition of a core utilizing thimble plugs.

Therefore, thermal hydraulic conditions which affect the structural integrity of the reactor vessel internals, such as reactor coolant temperatures and component lift forces, have been evaluated previously.

RCS thermal transients are not affected by the implementation of the TPC (Section 2.3.2), therefore, thermal stresses in the reactor vessel internals produced by transient thermal conditions need not be reevaluated.

Flow Induced Vibrations

As described in Section 2.3.3.2, the negligible change in the fuel assembly dynamic characteristics resulting from the implementation of TPBARs, combined with the fact that the mechanical design flow rates and design coolant temperatures do not change, will produce an insignificant change in loads associated with flow induced vibrations (FIV) for the reactor internals. Therefore, reevaluation of the reactor internals for revised flow induced vibration loads is not required and the existing component qualifications for FIV remain valid.

Gamma Heating

Based on the TPC equilibrium fuel cycle data (see Section 2.4.3), the magnitude of the radial leakage of neutrons and photons into the baffle/barrel region and the axial leakage to the lower core plate region falls between that observed with the design basis (CRSD) core power distribution and the Cycle 4 low leakage core power distribution implemented at the reference plant. In fact, from the standpoint of radial and axial leakage, the TPC equilibrium cycle is very similar to the reference plant Cycle 2 core power distribution. Since the actual core power distributions that have been implemented at the reference plant bracket the proposed TPC equilibrium cycle (i.e., Cycle 1 resulted in higher heating rates and Cycle 4 resulted in lower heating rates), the previous baffle/barrel and lower core plate heating rates bound the effects that are introduced by the implementation of the TPC core loading pattern defined in Section 2.4.3. Therefore, a detailed reevaluation of baffle/barrel and lower core plate heating rates is not warranted for the reference plant.

Given that the RCS thermal transients do not change significantly and the gamma heating rates of previous core loadings are bounding, the previous evaluations of thermal stresses in the baffle/ barrel and lower core plate are also bounding.

LOCA and Seismic

The conclusion of Section 2.3.3.1 is that the LOCA forces will not change with the implementation of TPBARs, thus reevaluation of the reactor vessel internals for the LOCA event is not required.

Since the dynamic characteristics of the fuel assembly remain unchanged (Section 2.4.2), the seismic accelerations realized by the RCS are not affected by the implementation of the TPC. Consequently, core plate motions and core interface loads during seismic events will not be impacted and existing seismic evaluations performed for previous cycle core loading remain valid.

2.3.6.7 Reactor Internals Structural Integrity Conclusions

The TPC has a negligible impact on the inputs to the reactor internals structural evaluation for the reference plant. Therefore, TPC implementation would have a negligible effect on the results and conclusions of previously performed reactor vessel internals structural integrity evaluations. Subsequently, evaluations which form the current design/operating basis for the reference plant remain valid for the TPC fuel pattern.

Since the TPC has a negligible effect on the loads applied to the reactor internals, changes in the loads transmitted to the reactor vessel from the reactor internals are also negligible.

2.3.6.8 Potential Plant Specific Analytical Issues

The reference plant reactor internals were not designed to the ASME Code. For a plant with internals designed per the ASME code to incorporate TPBARs, an increased structural evaluation effort would be required.

The reference plant does not have a thermal shield system. For a reactor with a thermal shield system to incorporate TPBARs, additional effort would be necessary to perform a structural and thermal evaluation of the thermal shield support system (bolts, flexures, and blocks). In the event that heating rates associated with previous qualified fuel cycles do not envelope the heating rates determined for the TPC, the thermal shield would need to be considered among the components to analyze for increased heating rates.

Given the grid load margin existing for the reference plant (see Section 2.4.2.1.5), no LOCA/seismic analysis was necessary. For a plant with significantly less grid load margin, a plant specific evaluation might require performing the LOCA/seismic analysis. This would involve assembling a plant-specific fuel assembly dynamic model to develop structural input (beam data) to use as input to the LOCA structural analysis.

For plant-specific implementation, the factors listed above (i.e., ASME Code internals, thermal shield system, grid load margin), as well as the impact of the TPC on the RCS parameters and NSSS design transients, should be reviewed to determine the impact on the reactor internals. As noted above, the plant-specific scope in this area has the potential to be significant.

2.3.7 EQUIPMENT QUALIFICATION (SRP 3.11) EVALUATION

2.3.7.1 Introduction

An evaluation was performed to determine the impact of the TPC on the environmental qualification (EQ) of mechanical and electrical equipment in the reference plant. The major impact on EQ for a TPC plant, as compared to a plant with a conventional core design, is the potential change in the radiation dose and dose rates resulting from differences in the core radiation sources. The impact of TPC on the equipment qualification doses was evaluated for several different accident and normal operating conditions, i.e.,

- Post-LOCA - TID 14844 Releases to Containment

- Post-LOCA - Recirculating Sump Water
- High Energy Line Break - Gap Activity Release to Containment
- Normal Operating Doses - Inside Containment
- Normal Operating Doses - Outside Containment

These conditions are consistent with SRP 3.11 and the Westinghouse methodology for qualification of safety related equipment as described in WCAP-8587 (Reference 1).

2.3.7.2 Evaluation

In general, the source terms associated with the longer lived nuclides are expected to be lower for a TPC plant than for current operating plants, since the TPC fuel assembly discharge burnups are lower than in conventional fuel cycles. This was confirmed by analysis, and it was further concluded that the dose rates and integrated doses considered in WCAP-8587 exceed the values associated with the TPC design.

As an example, in Figure 2.3.7-1, the calculated gamma dose rates in the middle of containment following a LOCA as presented in WCAP-8587 are compared to those calculated for a plant operating with a TPC. For conservatism, a small (ice condenser) containment volume is assumed for this comparison. As noted in the figure, the TPC plant dose rates are lower than the WCAP values by factors ranging from 1.13 - 2.24. The results are even more conservative for the reference plant, which has a dry containment. Since the free volumes of dry containments are more than a factor of two higher than ice-condenser containments and since the sources and doses are inversely proportional to containment volume, the reference plant (dry containment) TPC doses would be at least a factor of two lower than those presented in the figure. An additional conservatism in the analyses is the fact that no credit is taken for shielding that is provided by internal structures, such as shield walls and equipment.

The calculated post-LOCA beta doses inside a plant containment are presented in Figure 2.3.7-2. The beta dose rates as presented in WCAP-8587 are compared to the plant operating with a TPC. The TPC doses include the beta dose from tritium, conservatively assuming that the total end-of-life tritium activity inventory is distributed within the containment atmosphere, and are noted to differ from the WCAP values by factors ranging from 0.99 to 1.38. At short times after LOCA, the tritium dose rate contribution is at least two orders of magnitude lower than that from the fission products. Owing to the long half-life (i.e., 12.3 years) of tritium, the relative dose rate contribution increases such that the WCAP and TPC values are essentially the same at one year after a LOCA. However, since the WCAP-8587 dose rate values are generally not exceeded, the integrated doses at the post-LOCA time period of interest (i.e., less than one year) will be bounded by the WCAP values.

2.3.7.3 Conclusions

In summary, the evaluation concludes that the dose rates and integrated doses considered in WCAP-8587, which reflect the current Westinghouse methodology for equipment qualification of NSSS Safety Related Electrical Equipment, bound the values associated with the TPC design.

2.3.7.4 References

1. WCAP-8587, Revision 6-A (NP), "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment", G. Butterworth and R.B. Miller, March 1993.

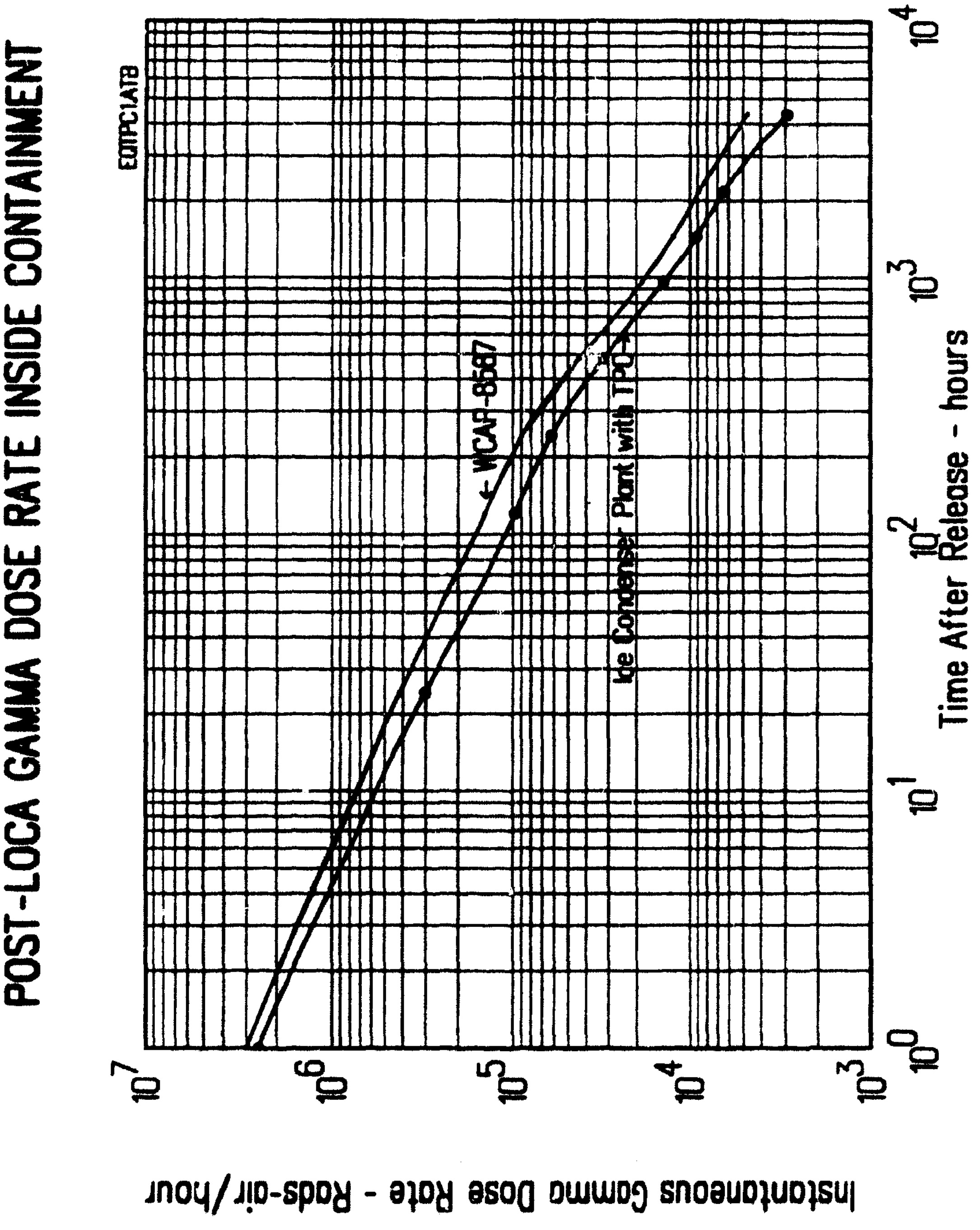

Figure 2.3.7-1
Post-LOCA Gamma Dose Rate Inside Containment

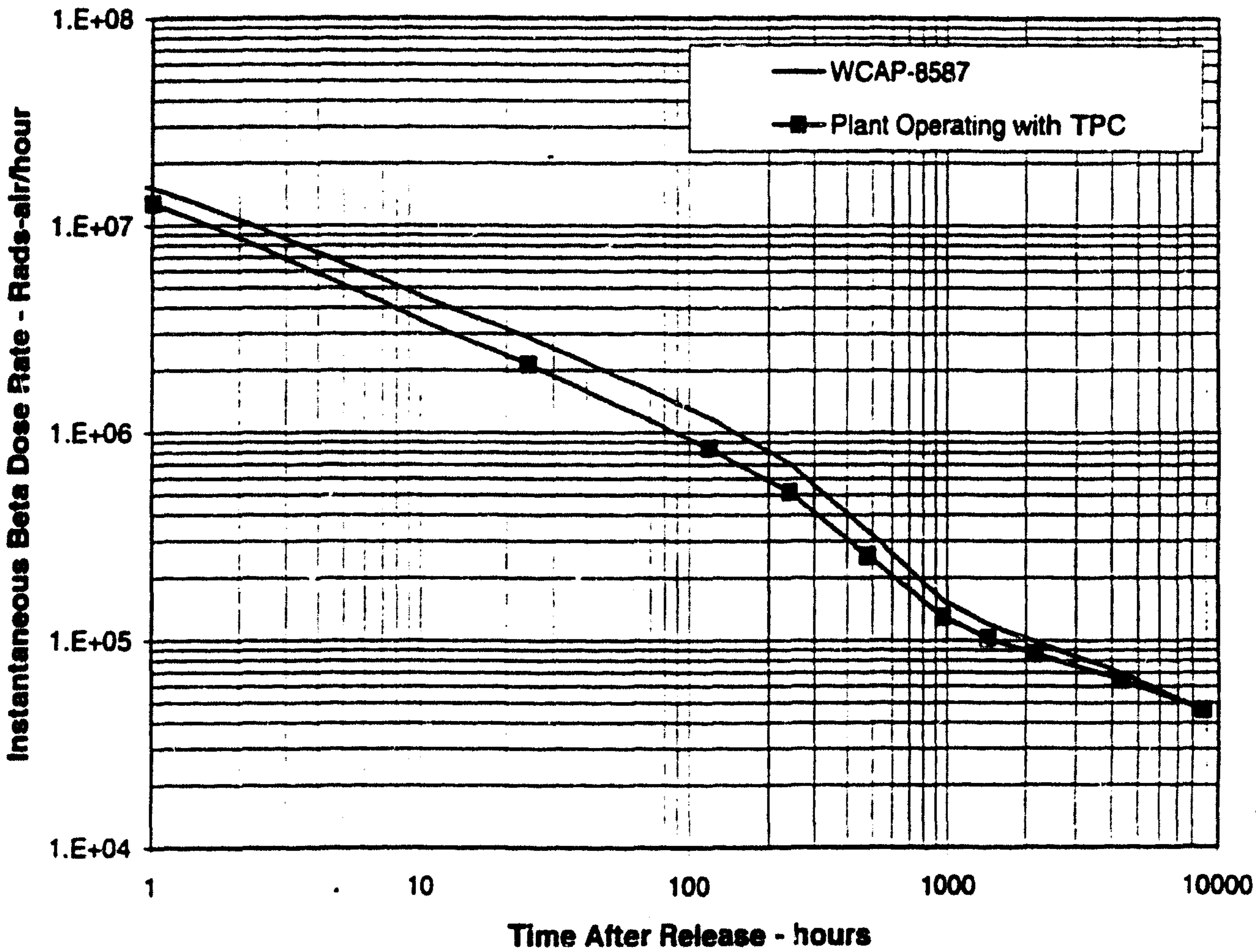

Figure 2.3.7-2
Post-LOCA Beta Dose Rate Inside Containment

2.4 REACTOR

2.4.1 INTRODUCTION

The sections in Chapter 4 of the SRP deal with the design of the reactor (including fuel design, nuclear design, thermal hydraulic design, and control rod drive system functional design) and the structural materials used in the control rod drive mechanism, reactor internals, and core support. Sections 2.4.2, 2.4.3 and 2.4.4 describe the evaluations of the impact of TPBARs on the fuel design, nuclear design, and thermal hydraulic design, respectively. Section 2.4.5 provides the evaluation of the impact of TPBARs on rod drop time.

SRP 4.2
Fuel System Design: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.46; 10 CFR 50, Appendix A, GDC 10, 27 and 35; 10 CFR 50, Appendix K; and 10 CFR 100. They specifically deal with the adequacy of the design bases, including fuel system damage, fuel rod failure, and fuel coolability; the adequacy of the fuel system description and design drawings; the acceptability of the design evaluation with respect to the design bases; and the adequacy of the testing, inspection, and surveillance plans. The objectives of the safety review of this section are to provide assurance that a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, b) fuel system damage is never so severe as to prevent control rod insertion when it is required, c) the number of fuel rod failures is not underestimated for postulated accidents, and d) coolability is always maintained. The impact of the incorporation of TPBARs into the fuel system design has been evaluated and is described in Section 2.4.2 of this document.

SRP 4.3
Nuclear Design: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 10, 11, 12, 13, 20, 25, 26, 27 and 28. They specifically deal with the acceptability of the core power distribution; the adequacy of the reactivity coefficients; the descriptions of the control requirements, control rod patterns and reactivity worths; and the acceptability of the analytical methods and data. Additional areas of review in this section include criticality calculations for single assemblies and groups of assemblies which are used in SRP sections 9.1.1 and 9.1.2, and the calculation of the neutron irradiation of the reactor vessel for use in SRP sections 5.3.2 and 5.3.3. The impact on the nuclear design from the incorporation of TPBARs into the core has been evaluated and is described in Section 2.4.3 of this document.

SRP 4.4
Thermal and Hydraulic Design: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 10. They specifically deal with the design bases for Departure from Nucleate Boiling (DNB), fuel temperature, core flow, and hydrodynamic stability; the applicability of the methodology used in determining that the design bases are met; the specification of appropriate tests and measurements; and the provision for appropriate instrumentation. The impact on the thermal and hydraulic design from the incorporation of TPBARs into the core has been evaluated and is described in Section 2.4.4 of this document.

SRP 4.5.1
Control Rod Drive Structural Materials: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a and 10 CFR 50, Appendix A, GDC 1, 14 and 26. They specifically deal with the properties of the materials used for the control rod drive mechanism, the control of austenitic stainless steel components, the compatibility of all materials used in contact with the reactor coolant, and the adequacy of onsite cleaning and cleanliness control procedures. The materials used in the control rod drive mechanism and the cleaning and cleanliness control procedures are not being changed for the TPC. The effects of the TPC on reactor coolant chemistry were evaluated (see Section 3.8.3 for details). In the worst case for normal operation, where two TPBARs are breached, the reactor coolant chemistry remains well within the design limits for impurities and particulates. Given that the reactor coolant chemistry is not significantly affected, and the materials have not changed, there is no impact due to the TPC on this SRP.

SRP 4.5.2
Reactor Internal and Core Support Materials: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a and 10 CFR 50, Appendix A, GDC 1. They specifically deal with the specification of materials to be used for major components of the reactor internals, the adequacy of welding controls for the internals, the nondestructive examination of wrought seamless tubular products and fittings, the control of austenitic stainless steel to prevent stress-corrosion cracking, and the compatibility of materials with the reactor coolant. The materials used in the reactor internals are not changed as a result of the TPC. The effects of the TPC on reactor coolant chemistry were evaluated (see Section 3.8.3 for details). In the worst case for normal operation, where two TPBARs are breached, the reactor coolant chemistry remains well within the design limits for impurities and particulates. Given that the reactor coolant chemistry is not significantly affected, and the materials have not changed, there is no impact due to the TPC on this SRP.

SRP 4.6
Functional Design of Control Rod Drive System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 23, 25, 26, 27, 28 and 29. They specifically deal with the capability of the Control Rod Drive System (CRDS) to perform its safety-related function following any assumed credible failure of a single active component, the capability to isolate essential portions from nonessential portions, the adequacy of the CRDS cooling system, the design of redundant reactivity control systems to preclude common mode failures, and the specification of appropriate functional tests. The functional design of the CRDS is not being changed as a result of the TPC. An evaluation of the impact of the TPBARs on RCCA rod drop time was performed and is discussed in Section 2.4.5 of this document.

2.4.2 FUEL DESIGN (SRP 4.2) EVALUATION

2.4.2.1 Fuel Assembly Structural Integrity

2.4.2.1.1 Introduction

A fuel assembly structural integrity evaluation has been performed to determine the impact of a TPC. The related sections of the Standard Review Plan with respect to fuel assembly integrity

have been reviewed to consider those factors where the TPBAR would affect the mechanical design. Only one structural factor, the weight of the fuel assembly containing 24 TPBARs, has changed with respect to the reference fuel assembly configuration and from previous SRP required analyses.

While the normal TPC TPBAR assembly consists of 24 TPBARs attached to a base plate, a few TPBAR assemblies will consist of TPBARs and either primary or secondary source rods. The combination of source rods and TPBARs on the same base plate was done to maximize the number of TPBARs in the reactor and hence the tritium production rate. The TPBAR/primary source assemblies each consist of 23 TPBARs and 1 primary source rod and the TPBAR/secondary source assemblies consist of 20 TPBARs and 4 secondary source rods. Primary source rods operate for one cycle and are removed from the reactor and not reused. The secondary source rods are irradiated for 10 to 15 cycles; during the first cycle they become activated and during the subsequent cycles they provide a source of neutrons. To facilitate removal of the source rods from the TPBAR base plate assemblies, and the reinsertion of secondary source rods in combination with unirradiated TPBARs, a design change to the base plates is envisioned for only those assemblies that contain source rods. One possible concept that would allow for the separation of the two types of rods for independent handling would be a partial plate system that could support TPBARs and be interlocked with another partial plate system holding source rods. Such a hardware change would facilitate the handling of TPBARs and source rods in the spent fuel pool and provide a means of combining the irradiated secondary source rods with unirradiated TPBARs with minimal personnel exposures. The specifics of the partial plate system design need to be developed once a specific plant is selected and the production requirements are better defined and the need for the partial plate system is known.

2.4.2.1.2 Methodology

An evaluation was performed to compare the impact of the additional weight of each fuel assembly to the grid load margin available for the reference plant in the fuel assembly structural analysis. The structural adequacy of the Westinghouse fuel assembly design is evaluated using NRC requirements for combined seismic and LOCA loads per Appendix A to SRP 4.2 and the approved methodology (Reference 1). The grid load results for the 17x17 VANTAGE+ fuel assembly design in the reference plant were reviewed. The combined seismic and LOCA grid load is less than [ ]^{a,c} of the allowable grid strength. Thus, the reference plant has more than [ ]^{a,c} grid load margin.

2.4.2.1.3 Input Parameters and Assumptions

Per Reference 2, the weight for each TPBAR in the lead test assembly (LTA) is 2.26 lbs. To allow for future design flexibility, 5% additional margin has been incorporated in the evaluations for the TPC, for a total weight allowance per TPBAR of approximately 2.37 lbs. Therefore, the additional weight per assembly totals approximately 62 lbs (including 24 TPBARs and 5.5 lbs for the holddown assembly). This is approximately 4% of a typical fuel assembly's weight.

2.4.2.1.4 Results

Because the TPBAR assembly is a hanging structure supported by the top nozzle adapter plate of the fuel assembly and the rodlets are hanging in the guide thimble tubes, the added weight

can be considered to be a part of the fuel assembly nozzle support. The added TPBAR assembly weight, together with the rodlet stiffness, has an insignificant effect on the fuel assembly's dynamic characteristics. Therefore, in plants that use the TPBAR assembly, there is no significant impact to the fuel assembly structural integrity evaluation. The design basis analyses/evaluations performed for the fuel assembly structural integrity assessment for the reference plant remain applicable.

2.4.2.1.5 Conclusions

The grid load margin for the reference plant has been reviewed. With a conservative load increase penalty factor of 20% (due to the added TPBAR weight effect) there is still more than sufficient grid load margin. Thus, the use of TPBAR assemblies in the reference plant has no impact on fuel assembly structural integrity. Furthermore, it is concluded that the LOCA and seismic combined loads will not be affected for the reference plant, for the full complement of TPBARs. This is because the LOCA/seismic methodology used for the reference plant permits the analyst to treat the TPBARs, because they are within the guide tubes, as part of the fuel assembly nozzle support. Dynamic characteristics of the fuel assembly are not affected. Interactions between the TPBARs and guide tubes tend to increase the fuel assembly damping properties. The range of motion of the TPBARs within the guide tubes is very limited, so that LOCA/seismic induced motion of the TPBARs is negligible. These factors reduce the impact of the added weight of the TPBAR assemblies on the LOCA/seismic analysis for the reference plant.

2.4.2.1.6 Applicability to other PWRs

For application of a full complement of TPBARs (24 TPBARs per fuel assembly) to other 3 and 4 loop PWRs, it is anticipated that the fuel assembly structural integrity and functional requirements will continue to be met, due to the grid load margin that exists in most PWRs. However, plant specific analyses will be required to confirm this conclusion. The interaction between the TPBARs and the host fuel assembly should also be considered in the seismic and LOCA analyses as appropriate with respect to the methodology used for the implementing plant. The results of these plant-specific evaluations provide input to the reactor internals and reactor vessel structural analyses.

2.4.2.1.7 References

1. WCAP-9401-P-A, "Verification Testing and Analyses of the 17x17 Optimized Fuel Assembly," Westinghouse Proprietary Class 2, August, 1981

2. PNNL-11419, Rev. 1, "Report on the Evaluation of the Tritium Producing Burnable Absorber Rod Lead Test Assembly," Pacific Northwest National Laboratory, March, 1997

2.4.2.2 Fuel Rod Design

2.4.2.2.1 Introduction

The primary function of a fuel rod is to generate and transfer heat to the reactor coolant. In the process of generating this heat via fissioning both radioactive and stable fission products are produced in the fuel. A second function of the fuel rod is to contain these fission products within the rod so that the reactor coolant does not become contaminated. To meet this goal the structural integrity of the fuel rod must be maintained (i.e., fuel damage or penetration of fuel rod clad is to be precluded).

The fuel rod must be capable of performing its intended function over a variety of operating conditions which occur during normal operation (Condition I) or are postulated to occur on a frequent basis (Condition II).

The integrity of the fuel rods during Condition I and Condition II modes of operation is ensured by proper fuel rod design. This is achieved by designing the fuel rods so that specific design criteria are satisfied for Condition I and II events. This is accomplished by demonstrating that the predicted performance of the limiting fuel rod with appropriate allowance for uncertainties is within the limits specified by each criterion. The performance evaluation considers the uncertainties associated with design models and the possible variations in as-built dimensions. Due to the empirical basis for the performance models (e.g., cladding creep, fission gas release, fuel densification, etc.) used in the design code, there is variability in the data used for model validation. Hence, it is necessary to account for deviations from the best estimate model projections to have confidence that the extremes of the performance spectrum are covered. In the conservative case evaluation, a statistical convolution is typically performed from the individual deviations in a performance parameter caused by these model and dimensional uncertainties. The overall uncertainty in a particular performance parameter is calculated as the root mean square of the individual effects. The performance parameters are those predicted quantities which serve as indicators of fuel rod behavior. These performance parameters include rod internal pressure, fuel temperature, cladding stress and cladding strain.

Extremely low probability occurrences which have the potential to cause significant fuel damage are classified as Condition III and Condition IV events. Fuel rod integrity cannot be guaranteed during these hypothetical occurrences. For these occurrences, a failure analysis is conducted as part of the plant safety analysis for each of these events and offsite dose calculations are performed to confirm that regulations on radioactive release are satisfied.

The incorporation of the TPBARs in the core alters the fuel management and subsequently the duty of the fuel relative to a design without TPBARs. This change in fuel duty has been assessed to determine if all fuel rod design criteria can be satisfied.

2.4.2.2.2 Methodology

Design models used in the evaluation of fuel rod design criteria have been licensed by the NRC for design applications up to a lead rod average burnup of 60,000 MWD/MTU (Reference 1).

The NRC-approved PAD 3.4 code, with NRC-approved models for in-reactor behavior (References 2 and 3), 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, cladding elastic and plastic behavior, fission gas release, and fuel densification and swelling as a function of time and linear power.

PAD 3.4 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 as-built dimensional tolerances is used. Each model which has a significant effect on rod performance includes appropriate uncertainty bands defined to bound 95% of the data. These uncertainty bands are used to define conservative upper bound uncertainty levels in the model predictions. Consideration of these uncertainty levels in the conservative case evaluation for the performance of the limiting rods within a fuel region provides adequate assurance of meeting the design goal that all fuel rods in that region satisfy the design criteria. The uncertainty in rod performance associated with variations in the pellet, cladding and rod dimensions and fuel characteristics such as density are also considered. The overall uncertainty in a particular performance parameter is calculated as the root mean square of the individual effects. As-built dimensional uncertainties have been measured for some critical inputs, e.g., fuel pellet diameter, and where available, these fabrication uncertainties are used in place of the manufacturing tolerances.

The fuel rod design criteria given in Section 2.4.2.2.4 are verified by evaluating the predicted performance of the limiting fuel rod, defined as the rod which gives the minimum margin to the design limit. In general, no single rod is limiting with respect to all the design criteria. Generic evaluations have identified which rods are most likely to be limiting for each criterion, and these rods are evaluated to determine if each of the design criteria are met.

Fuel rod design evaluations for the Tritium Production Core (TPC) fuel were performed using the NRC-approved models in References 2 and 3, and standard design methods to demonstrate that all fuel rod design criteria (References 1 and 4) are satisfied.

The TPC fuel rods are evaluated using standard design methods by modeling the core operating conditions, fuel rod power duty, and axial power shapes from their initial insertion into the core through the end of their irradiation. During this depletion, the changes in fuel rod geometry due to cladding creep, rod axial growth, fuel densification and swelling are accounted for along with the fuel temperatures and fission gas release effects. During the depletion, Condition II events are simulated to account for the increase in fission gas inventory due to transient fission gas release and to evaluate the impacts of the Condition II events on cladding stress and strain.

2.4.2.2.3 Significant Input Parameters and Assumptions

The specific assumptions used in the verification of the fuel rod design criteria for the TPC fuel include: (1) TPC specific operating conditions (core power, flow rate, inlet temperature, system pressure), and (2) fuel rod duty (steady state rod powers, axial power shapes, Condition II local rod powers, etc.).

The fuel rod design parameters for TPC are essentially the same as a standard Westinghouse 17X17 fuel rod with the following specifications:

Fuel Rod Design Parameter Comparisons		
Parameter	**TPC**	**Reference Core**
H0X Enrichment, w/o (High Enrichment, Fresh Fuel)	4.95 (196 Rods/Assembly) 4.15 (68 Rods/Assembly)	4.20 (264 Rods/Assembly)
L0X Enrichment, w/o (Low Enrichment, Fresh Fuel)	4.60 (196 Rods/Assembly) 3.45 (68 Rods/Assembly)	4.00 (264 Rods/Assembly)
IFBA Parameters: B-10 Loading, mg/in B-10 Coating Length, in	 1.50 128.5	 1.88 [ ]a,c
Helium Backfill Pressure: IFBA Rod, psig Non-IFBA Rod, psig	 [ ]a,c [ ]a,c	 [ ]a,c [ ]a,c
Plenum Length, in	[ ]a,c	[ ]a,c
Cladding Material	ZIRLO™	ZIRLO™

2.4.2.2.4 Acceptance Criteria and Results

To ensure reliable operation, fuel rod design criteria have been established for all operating conditions consistent with Condition I and/or Condition II events. The criteria pertinent to the fuel rod design are as follows:

- Rod Internal Pressure
- Cladding Stress
- Cladding Strain
- Cladding Oxidation and Hydriding
- Fuel Temperature
- Cladding Fatigue
- Clad Flattening
- Fuel Rod Axial Growth

Each of these key fuel rod design criteria have been evaluated for the TPC fuel rods. Based on these evaluations, it is concluded that the fuel rod design criteria are satisfied for the TPC fuel duty. The design criteria and results are described in more detail below.

Rod Internal Pressure

The internal pressure of the lead fuel rod in the reactor will be limited to a value below that which could cause (1) the diametral gap to increase due to outward cladding creep during steady state operation or (2) extensive DNB propagation to occur.

The rod internal pressure for the TPC fuel rods have been evaluated by modeling the fuel rod power duty, axial power shapes of the fuel rods from their initial insertion into the core through to their end of irradiation. The rod internal pressure criterion allows the internal pressure to exceed the system pressure. The calculated rod internal pressure is compared to the allowable rod internal pressure at each depletion step where the internal pressure exceeds system pressure. The allowable rod internal pressure or pressure limit is dependent upon the cladding temperature and internal pressure which vary during depletion. The rod internal pressure analysis has shown that the upper bound rod internal pressure was less than [ ]^{a,c} and that the gap reopening portion of the rod internal pressure criterion was satisfied for the TPC fuel rod design.

The second part of the rod internal pressure design basis precludes extensive DNB propagation and associated fuel failure. The basis for this criterion is that no significant additional fuel failures will occur in cores which have fuel rods operating with rod internal pressure in excess of system pressure due to DNB propagation. The design limit for Condition II events is that DNB propagation is not extensive, i.e., the process is shown to be self limiting and the number of additional rods in DNB due to propagation is relatively small. For Condition III and IV events, it is shown that the total number of rods in DNB, including propagation effects, is consistent with the assumptions used in radiological dose calculations for the event under consideration. The evaluations performed for the TPC fuel rod design have shown that there will be no extensive DNB propagation.

Cladding Stress

The design limit for clad stress is that the volume average effective stress considering interference due to uniform cylindrical pellet-cladding contact caused by pellet thermal expansion, pellet swelling and uniform cladding creep, and pressure differences is less than the cladding yield strength under Condition I and II events.

While the cladding has some capability for accommodating plastic strain, the yield stress has been established as the conservative design limit. Cladding temperature and irradiation effects on the yield strength are considered.

The cladding stress calculated for the fuel rods in the TPC showed sufficient margin, [ ]^{a,c}, to the cladding yield strength. Thus the cladding stress criterion is met.

Cladding Strain

The total plastic tensile creep strain due to uniform cladding creep and uniform cylindrical fuel pellet expansion associated with fuel swelling and fuel thermal expansion is less than 1% from the unirradiated condition. The acceptance limit for fuel rod clad strain during a Condition II

event is that the total tensile strain due to uniform cylindrical pellet thermal expansion is less than 1% from the pre-transient value.

The fuel rod design and core operating conditions for the TPC were shown to be bounded by the assumptions made for a generic bounding steady-state cladding strain analysis performed for the Westinghouse 17x17 fuel rod. The generic analyses showed that the steady-state strain for rod average burnups in excess of 60,000 MWD/MTU was less than the 1% end-of-life strain design limit. Thus, the TPC fuel rod design meets the 1% steady-state strain limit.

For Condition II events, the total tensile strain was less than 1% from the pre-transient value.

Oxidation and Hydriding

The calculated ZIRLO™ cladding temperature (metal-oxide interface temperature) will be less than [ ]^{a,c} during steady-state operation. For Condition II events, the calculated cladding temperature will not exceed [ ]^{a,c}. The hydrogen pickup level in ZIRLO™ cladding and structural components shall be no greater than [ ]^{a,c} at end-of-life.

The limitation on the cladding temperature minimizes the potential for accelerated cladding corrosion. The hydrogen pickup criterion limits the loss of ductility due to hydrogen embrittlement which occurs upon the formation of zirconium hydride platelets.

The corrosion analysis has shown that the cladding steady-state and transient temperatures were less than [ ]^{a,c}, respectively, for the fuel rods in the TPC. The hydrogen pickup was less than [ ]^{a,c} for both the cladding and the structural components. Thus, the oxidation and hydriding criteria are satisfied for the TPC.

Fuel Temperature

The Westinghouse design limit for fuel temperatures during Condition I and II events is that there is at least a 95% probability that the calculated centerline fuel temperature does not exceed the fuel melting temperature. The melting temperature of unirradiated UO_2 is taken as 5080°F, decreasing 58°F per 10,000 MWD/MTU exposure. A fuel centerline temperature limit of 4700°F has been selected by Westinghouse as the design limit. The difference between this 4700°F value covers the model and manufacturing uncertainties and the allowance for the reduction in melting temperature with burnup.

The intent of this criterion is to avoid a condition of gross fuel melting which can result in severe duty on the cladding. The concern here is based on the large volume increase associated with the phase change in the fuel and the potential for loss of cladding integrity as a result of molten fuel/cladding interaction.

For the TPC fuel rods, the local linear power remains below the value which will result in fuel centerline melt.

Clad Fatigue

The fuel rod design criterion for cladding fatigue requires that the fatigue cumulative usage factor is less than 1.0. That is, for a given strain range, the number of strain fatigue cycles are less than those required for failure, considering a factor of safety of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles, whichever is more conservative.

The concern of this criterion is the accumulated effect of short-term cyclic, cladding stress, and strain, which result from daily load follow operation.

The evaluation of cladding fatigue limit assumed daily load follow over the life of the fuel rod. The accumulated effects of cyclic strains associated with normal plant shutdowns and returns to full power were also considered. The TPC fuel rod fatigue evaluation showed that the fatigue cumulative usage factor was [ ]^{a,c}, which is well below the design usage factor limit of 1.0.

Clad Flattening

Clad flattening shall not occur during the projected exposure lifetime of the fuel.

This clad flattening criterion prevents fuel rod failures due to long-term creep collapse of the fuel rod cladding into axial gaps formed within the fuel stack due to fuel densification.

In Reference 4, Westinghouse has presented post irradiation and in-core flux trace data which confirm that significant axial gaps in the fuel column due to densification will not occur for current Westinghouse fuel designs. These current fuel rod designs employ fuel with improved in-pile stability which provide adequate assurance that axial gaps large enough to allow clad flattening will not form within the fuel stack.

The TPC fuel design is a current fuel design with respect to the parameters (cladding creep, fuel densification, initial gas pressure and pellet-clad gap) which affect the pellet-clad gap behavior. Therefore, the Reference 4 report applies to the fuel in the TPC and clad flattening will not occur.

Fuel Rod Axial Growth

The design limit for fuel rod growth is that no interference between the fuel rods and the fuel assembly top and bottom nozzles will occur.

This criterion assures that sufficient axial space exists to accommodate the maximum expected fuel rod growth without degradation of the fuel rod or fuel assembly function. Fuel rods are designed with adequate clearance between the fuel rod and the top and bottom nozzles to accommodate the differences in the growth of fuel rods and the growth of the fuel assembly.

The TPC fuel rod growth evaluation has shown that the fuel rod can achieve rod average burnups in excess of 60,000 MWD/MTU rod average burnup without violating the rod growth criterion.

2.4.2.2.5 Conclusions

All fuel rod design criteria are met for the TPC design. The use of TPBARs has no significant impact on meeting the fuel rod design criteria. The fuel duty experienced by the fuel rods in the TPC design was not significantly different from typical reload design and the resulting fuel rod performance parameters of rod internal pressure, cladding corrosion, cladding stress, cladding strain and cladding fatigue were similar to those seen in non-TPC reload designs. Since the fuel design is standard and standard design criteria are met, no fuel rod related-documents need to be modified for the TPC fuel.

2.4.2.2.6 Potential Plant Specific Analytical Issues

Fuel rod design analysis of rod internal pressure, corrosion, cladding stress, strain and fatigue are typically performed for each reload cycle utilizing plant and cycle specific data related to operating conditions and fuel rod duties. The fuel rod design analyses performed for the TPC design have shown margins to the design criteria limits such that implementation in other plants would also be expected to show acceptable results.

2.4.2.2.7 References

1. WCAP-10125-P-A, "Extended Burnup Evaluation of Westinghouse Fuel," Davidson, S. L., et al, December 1985.

2. WCAP-11873-A, "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," Weiner, R. A., et al., August 1988.

3. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," Davidson, S. L., Nuhfer, D.L., April 1995.

4. WCAP-13589-P-A, "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," Kersting, P. J. et al., March 1995.

2.4.3 NUCLEAR DESIGN (SRP 4.3)

2.4.3.1 Introduction

First and equilibrium cycle conceptual core designs have been developed for the Tritium Production Cores (TPCs). The overall goal of these conceptual core designs and their associated analyses was to establish the feasibility of using a typical PWR for large scale production of tritium. A representative Westinghouse four-loop plant was chosen as the reference reactor (Reference 1), and core designs were developed to meet typical cycle energy goals. The cycle energy chosen for the designs was 494 Effective Full Power Days (EFPD). This is equivalent to an 18 month cycle at a capacity factor of approximately 90%. The cycle burnup that corresponds to this energy is 21,564 MWD/MTU. This cycle energy assumption is representative of typical PWR fuel cycles.

From a nuclear design viewpoint, the Tritium Producing Burnable Absorber Rods (TPBARs) function in the reactor core in a manner similar to standard Pyrex burnable absorbers or wet annular burnable absorbers (WABAs) (Reference 2). They are effective in holding down initial excess core reactivity and shaping the core power distribution. In terms of reactivity worth, at beginning-of-life (BOL) TPBARs are worth slightly less than WABAs or Pyrex burnable absorbers, depending upon the linear loading of lithium-6 employed. The primary design goal for these cores was to maximize the production of tritium. To that end, the designs use the maximum number of TPBARs possible. The feed region sizes and enrichments were varied to achieve the desired cycle energy. As a consequence of this design goal, the fuel economy of these core designs is less efficient than typical core designs. The use of TPBARs necessarily increases fuel cycle costs because of the large reactivity penalty of the TPBARs at end-of-life (EOL). This large reactivity penalty occurs since, unlike conventional burnable absorbers, TPBARs do not deplete completely. This is a direct result of the large initial loading of lithium-6 (^{6}Li), the key TPBAR neutron absorber, and the (relatively) low neutron absorption cross section of ^{6}Li (compared to boron-10, for example).

Table 2.4.3-1 lists some of the TPC operating parameters and design objectives. Both the first cycle and equilibrium cycle core designs use the same fuel type--Westinghouse 17x17 VANTAGE+ fuel (Reference 3) with IFM grids and ZIRLO™ fuel cladding. This is the same fuel type employed in the reference plant. As much as practical, the cores were designed to have power distributions, peaking factors, and critical boron concentrations that are comparable to those routinely observed in the reference plant. Also, the cores were designed to meet established design and safety limits. For example, the $F_{\Delta H}{}^N$ and $F_Q{}^T$ limits for the reference plant are 1.65 and 2.50, respectively. The moderator temperature coefficient limit at hot zero power (HZP) is +7 pcm/°F. The shutdown margin (SDM) limit is 1.3 %Δρ. A comprehensive set of nuclear design analyses has been completed for these cores in which key safety parameters were calculated and compared to the values assumed in the reference plant safety analyses. The approved methodology employed to do this is described in Reference 4. The result of these analyses is that, with only minor exceptions that do not affect the conclusions, the key safety parameters for these cores fall within the ranges that are typically assumed for the reference plant.

The primary difference between the TPC designs and typical reload cores is the use of TPBARs. As stated above, a key objective of these designs was to employ the maximum number of TPBARs possible while still meeting design and safety limits. Accordingly, the conceptual designs employ a cluster of TPBARs in every possible core location (every assembly except the RCCA locations). Reference 5 provides a detailed description of the TPBAR design used in the Watts Bar Lead Test Assemblies (LTAs). The TPBAR design used here is neutronically the same as the LTA design except that the linear loading of ^6Li and the active absorber length are assumed to be design variables. In these TPC designs, the ^6Li linear loading used is 0.030 gm/in, as opposed to 0.0247 gm/in the LTAs. The active length of the TPBARs is 127.5 inches for the TPC first cycle and 128.5 inches for the equilibrium cycle, whereas the LTA active length was 142 inches. While the active length is smaller for the TPBARs in these designs, the higher linear loading increases tritium production, so that the grams of tritium produced per TPBAR are roughly comparable to the LTA for TPBARs loaded into the core interior. A shorter active length was used to help enhance the axial power distribution shape. The use of "part-length" burnable absorbers in this fashion is common practice in conventional core designs.

Several other differences between these TPC designs and typical core designs are worth noting. First, enrichment zoning was employed in the fuel assemblies using TPBARs. Specifically, the outer row of fuel pins (and an additional rod in each corner of the fuel assembly) was given lower ^{235}U enrichments than the interior fuel pins. (The terms "fuel rod" and "fuel pin" are synonymous in this report.) This was done to shape the intra-assembly power distribution, lower the power peaking, and increase slightly the total tritium production in the TPBARs.

Second, larger feed regions and slightly higher fuel enrichments were used than is typical of current reference plant core designs. For example, the feed region size of the equilibrium cycle is 140 assemblies. Typically, a four-loop plant will feed 84-92 fuel assemblies each cycle. Large feed regions and high enrichments are necessary for the TPCs because of the large residual reactivity penalty of the TPBARs mentioned above. This penalty requires a higher initial excess core reactivity to achieve the desired cycle energy. The higher initial core reactivity takes the form of more feed assemblies and higher enrichments. A maximum enrichment of 4.95 w/o ^{235}U was assumed. The high enrichment pins in some of the feed assemblies used 4.95 w/o in both the first and equilibrium cycle TPC designs.

Lastly, in addition to high loadings of TPBARs, large numbers of Integral Fuel Burnable Absorber (IFBA) fuel pins were needed in these designs. IFBA (Reference 6) fuel rods utilize a thin coating of ZrB_2 on the fuel pellet which serves to reduce the initial core reactivity and shape the power distribution, much like a discrete burnable absorber. The need for IFBAs is related to the high initial reactivity loading referred to above. The large number of feed assemblies combined with high enrichments require large loadings of IFBA fuel pins to help hold down initial excess core reactivity and maintain reasonable critical boron concentrations at BOL, and achieve an acceptable value for the moderator temperature coefficient.

Table 2.4.3-2 provides a list of symbols, terms, and abbreviations used throughout this section of the report.

2.4.3.2 Methodology

The primary analytical models employed to develop and analyze these core designs are described in References 7, 8, and 9. The key neutronics codes employed were the PHOENIX-P (Reference 7 and 8) and ANC (Reference 9) codes, which have been approved for use by the NRC.

PHOENIX-P is a two-dimensional, multi-group transport theory code which utilizes a 42 energy group cross section library. It provides the capability for cell lattice modeling on an assembly level. PHOENIX-P is used to provide homogenized, two-group cross sections for nodal calculations and feedback models. Specifically with respect to burnable absorbers, PHOENIX-P is used routinely to calculate macroscopic group constants for reactor cores which employ discrete burnable absorbers similar to TPBARs, absorbers such as the wet annular burnable absorber and the standard Pyrex burnable absorber.

While the methodology and solution algorithms of PHOENIX-P are applicable to TPBARs, the production version of PHOENIX-P did not include the necessary library cross sections for neutronic modeling or the necessary geometry logic for depletion of TPBARs. Consequently, to permit TPBAR modeling, minor code and library modifications were necessary. The basic solution algorithm of PHOENIX-P, which is based on collision probability and discrete ordinates methods, was specifically not modified. Just as in the case of WABAs or Pyrex burnable absorbers, these techniques are used, without modification, in the transport solution for assemblies containing TPBARs.

The TPBARs employ a lithium-aluminate pellet as the active neutron absorber. The primary neutron absorption reaction is:

$$^{6}\text{Li}\ (n,\alpha)\ \text{T}$$

The tritium product, T, has a 12.33 year half-life and decays into helium-3. Therefore, a secondary neutron absorption reaction is:

$$^{3}\text{He}\ (n,p)\ \text{T}$$

The lithium aluminate pellet also contains a large amount of ^{7}Li. Consequently, there are four principal isotopes which need to be modeled: ^{6}Li, ^{7}Li, ^{3}He, and T. None of these isotopes were in the standard PHOENIX-P library. Therefore, it was necessary to add cross section data to the library for these isotopes, and this was done using the basic nuclear data for these isotopes from ENDF-B/VI (Reference 10) and processing the data into the 42 energy group format required by PHOENIX-P. No other isotopes were added, and no other changes to the library were made, so that the library data for the standard set of isotopes is identical to that which is used routinely.

PHOENIX-P uses "pin-types" to distinguish different kinds of pin geometries in the lattice. For example, PIN TYPEWABA is the PHOENIX-P pin-type associated with wet annular burnable absorber geometry. PIN TYPEPYRX designates a standard Pyrex burnable absorber. The simple entry PIN TYPE designates fuel pin geometry. To model a TPBAR in PHOENIX-P, a

new set of geometry characteristics was added to the code logic under the designation PIN TYPELBAR. When PHOENIX-P encounters this entry in problem input, it knows to expect geometry and number density information associated with a TPBAR.

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As mentioned above, no changes were made to the code logic associated with the basic solution algorithm. Thus, PHOENIX-P will perform the neutron transport calculations for the TPBAR just as it would for a WABA or Pyrex burnable absorber. The testing of this version of PHOENIX-P is discussed in References 11 and 12.

To distinguish it from the version of PHOENIX-P which is used internally by Westinghouse for core analysis and which is transmitted to Westinghouse's technology transfer customers, the version of PHOENIX-P which includes the TPBAR modeling capability has been designated PHOENIX-L. Since the fundamental solution algorithm and cross section library are unchanged in this code version (except for the addition of the principal TPBAR isotopes and chains and TPBAR geometry modeling mentioned above), References 7 and 8 remain appropriate references for this code version. At this time there are no plans to incorporate TPBAR modeling capability in the general version of the code. Any future modifications of the TPBAR modeling capability will be made to the PHOENIX-L version and will be tested using the same process outlined in References 11 and 12, and accepted in Section 2.1.2 of the Safety Evaluation transmitted in Reference 18.

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In addition to modifications to PHOENIX-P, modifications to the reactor core model, ANC, were required for full TPBAR modeling capability.

ANC (References 7 and 9) is an advanced nodal code capable of two-dimensional and three-dimensional reactor core calculations. PHOENIX-P is used to calculate the necessary fuel and burnable absorber cross section data for ANC. ANC uses this data to predict core power distributions, reactivity coefficients, etc.

In order to explicitly model the tritium/helium-3 chain in ANC, it was necessary to add new code logic and data structures to ANC to permit the tracking and storing of the tritium and helium-3 isotopes on a node-by-node basis. By adding the tritium/helium-3 depletion chain equations to ANC, local (i.e., nodewise) depletion of both lithium-6 and helium-3 was made possible. The chain equations added to ANC for this purpose are identical to the chain equations programmed into PHOENIX-L.

The ANC code modifications described above have been incorporated into a version of ANC called ANC-L to distinguish it from the standard production version of ANC. No changes to the basic ANC neutronics solution algorithm were made in this code version. The changes to the code were limited to those required for explicit modeling of the TPBARs, as described above, and editing routines that provide convenient TPBAR modeling and performance information for the code user. Since no changes were made to the ANC fundamental solution algorithm, References 7 and 9 remain appropriate references for this code version. The testing of this version of ANC is described in Reference 12, and accepted in Section 2.1.2 of the Safety Evaluation transmitted in Reference 18.

In addition to the PHOENIX-P and ANC codes, the APOLLO code was used for some of the nuclear design analyses performed for the Tritium Production Cores. APOLLO (Reference 14), an updated version of the PANDA code (Reference 15), is a two-group, one-dimensional diffusion depletion code. It uses cross sections generated by radial averaging of the corresponding 3D model cross sections. APOLLO is routinely used for computing axial power distributions, differential rod worths, control rod operating limits, etc. For the TPC designs, APOLLO models were derived from ANC-L models. ANC-L provides the necessary information for APOLLO to model TPBARs. Consequently, no modifications to the standard version of APOLLO were required.

2.4.3.3 Design Bases

The design bases and functional requirements used in the nuclear design of the fuel and reactivity control systems for the TPC designs are the same as those used for the reference plant. The design bases and functional requirements are discussed in detail in Section 4.3 of Reference 1. This information is applicable to the TPC designs. Provided below is a discussion of these design bases and their relationship to TPBARs and the TPC designs.

Fuel Burnup

Basis:
A limitation on initial installed excess reactivity or average discharge burnup is not required

other than as is quantified in terms of other design bases, such as core negative reactivity feedback and shutdown margin.

Discussion:
As described in the introduction, the initial excess reactivity of the TPC designs is necessarily larger than typical for the reference plant in order to compensate for the residual reactivity penalty of the TPBARs at EOL. This initial excess reactivity is effectively controlled by the combined worths of the integral fuel burnable absorbers, the soluble boron in the reactor coolant (chemical shim concentration), and the TPBARs themselves. In this sense, the TPBARs function like conventional Pyrex burnable absorbers or WABAs. Relative to discharge burnup, the fuel average discharge burnup for the TPC designs is smaller than typical for the reference plant because of the large feed region size. The average discharge burnup for the equilibrium cycle TPC is less than 30000 MWD/MTU. For the reference plant, average discharge burnups of about 45000 MWD/MTU are typical.

Negative Reactivity Feedbacks (Reactivity Coefficients)

Basis:
The design basis for negative reactivity feedbacks specifies that the fuel temperature coefficient will be negative, and the moderator temperature coefficient (MTC) of reactivity will be non-positive for full power operating conditions, thereby providing negative reactivity feedback characteristics at full power. For the reference plant, a moderator temperature coefficient of up to +7.0 pcm/°F is allowed from 0% to 70% power. From 70% to 100% power, the moderator temperature coefficient limit decreases linearly from +7.0 to 0.0 pcm/°F.

Discussion:
For the TPC designs, the Doppler feedback is always negative and is roughly comparable to the Doppler feedback normally observed in the reference plant (see Section 2.4.3.6). The moderator temperature coefficients for TPC designs meet the MTC limits described above. In general, the TPC designs have moderator temperature coefficients at BOL which are slightly more negative than recent core designs for the reference plant. This is due, in part, to the slightly lower soluble boron worth in the TPC designs. At EOL, the MTCs are slightly less negative. The total power coefficient for the TPC designs is always negative at all power levels. Section 2.4.3.6 provides typical values of the reactivity coefficients for the TPC designs.

Based on the above, the design basis for reactivity feedback and the reactivity feedback behavior of the TPC designs meet General Design Criteria (GDC) 11 of 10 CFR 50, Appendix A.

Control of Power Distributions

Basis:
As stated in Reference 1, the nuclear design basis for control of power distributions is that, with at least a 95% confidence level:

A) The fuel will not be operated at greater than 14.5 kW/ft under normal operating conditions, including an allowance of 2 percent for calorimetric error.

B) Under abnormal conditions, including the maximum overpower condition, the fuel peak power will not cause melting as defined in paragraph 4.4.1.2 of Reference 1.

C) The fuel will not operate with a power distribution that violates the departure from nucleate boiling (DNB) design basis (i.e., the departure from nucleate boiling ratio (DNBR) shall not be less than the design limit DNBR discussed in subsection 4.4.1 of Reference 1) under Condition I and II events, including the maximum overpower condition.

D) Fuel management will be such as to produce values of fuel rod power and burnup consistent with the assumptions in the fuel rod mechanical integrity analysis of Section 4.2 of Reference 1.

Discussion:
The power distributions for the TPC designs were analyzed to ensure that the above design basis was met. The reference plant Technical Specifications place limits on power peaking (F_Q and $F_{\Delta H}$) and axial flux difference (AFD) for Condition I operation. Using the standard design methods of References 4 and 16, a range of radial and axial power distributions were calculated and analyzed for the TPC designs to ensure that the reference plant power peaking factor limits and DNB design basis were met. Relaxed Axial Offset Control (RAOC), as described in Reference 16, was assumed. Condition II power shapes were analyzed as well to demonstrate that the overpower kW/ft limit of 22.4 was met. This ensures that fuel melting criteria are satisfied.

Section 2.4.3.5 provides illustrative power distributions for the TPC designs.

The above basis is consistent with GDC 10 of 10 CFR 50, Appendix A. The analysis results confirm that GDC 10 is satisfied for the TPC designs.

Maximum Controlled Reactivity Insertion Rate

Basis:
The maximum reactivity insertion rate due to withdrawal of RCCAs at power or by boron dilution is limited. During normal at power operation, the maximum controlled reactivity insertion rate is limited. A maximum reactivity change rate for accidental withdrawal of two control banks is set so that peak heat generation rate and DNBR do not exceed the maximum allowable at overpower conditions. This satisfies GDC 25 of 10 CFR 50, Appendix A.

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited to preclude rupture of the coolant pressure boundary or disruption of the core internals to a degree which would impair core cooling capacity due to a rod withdrawal or an ejection accident (see Section 2.15).

Following any Condition IV event (rod ejection, steam line break, etc.), the reactor can be brought to the shutdown condition, and the core will maintain acceptable heat transfer geometry. This satisfies GDC 28 of 10 CFR 50, Appendix A.

Discussion:
The discussion of Reference 1 applies to the TPC designs. Reactivity addition associated with an accidental withdrawal of a control bank (or banks) is limited by the maximum rod speed (or travel rate) and by the worth of the bank(s). For the reference plant, the maximum control rod speed is 45 in/min.

The reactivity change rates are conservatively calculated, assuming unfavorable axial power and xenon distributions. The peak xenon burnout rate is significantly lower than the maximum reactivity addition rate for normal operation and for accidental withdrawal of two banks.

Shutdown Margins

Bases:
Minimum shutdown margin, as specified in the Technical Specifications or Technical Requirements Manual, is required in all operating modes, in the hot standby shutdown condition, and in the cold shutdown condition.

In all analyses involving reactor trip, the single, highest worth rod cluster control assembly is postulated to remain untripped in its full-out position (stuck rod criterion). This satisfies GDC 26 of 10 CFR 50, Appendix A.

When fuel assemblies are in the pressure vessel and the vessel head is not in place, k_{eff} will be maintained at or below 0.95 with control rods and soluble boron. Further, the fuel will be maintained sufficiently subcritical that removal of all rod cluster control assemblies will not result in criticality.

Discussion:
The discussion provided in Reference 1 for these bases applies to the TPC designs. Section 2.4.3.7 provides a shutdown margin assessment and demonstrates that the TPC designs will meet the 1.3 $\%\Delta\rho$ shutdown margin requirement for Modes 1 and 2 as given in the reference plant Technical Specifications. This assessment assumes the highest worth control rod is stuck out upon trip. Shutdown margin requirements for all operating modes can be met for the TPC designs. The boron concentration required to meet the refueling shutdown criteria is calculated using standard methods and is specified in the plant Technical Specifications.

Stability

Basis:
The core will be inherently stable to power oscillations at the fundamental mode. This satisfies GDC 12 of 10 CFR 50, Appendix A.

Spatial power oscillations within the core with a constant core power output, should they occur, can be reliably and readily detected and suppressed.

Discussion:
The discussion provided in Reference 1 applies to the TPC designs. No changes to the reactor control and protection systems are required for the TPC designs, so that suppression of and

protection against power oscillations at the fundamental mode will occur just as for typical reference plant core designs. While no specific xenon stability studies were performed as part of this study, azimuthal and diametral oscillations due to xenon effects are expected to be self-damping for the TPC designs since the reactivity feedbacks of the TPC designs are comparable to typical reference plant core designs.

However, just as for the reference plant core designs, axial xenon oscillations may occur. Such oscillations can be readily detected using the excore detectors and controlled using the control banks.

Anticipated Transients Without Scram

The effects of anticipated transients with failure to trip are not considered in the design bases of the plant. In general, the TPC designs will respond better to ATWS events than typical reference plant core designs due to slightly greater moderator feedback (See Section 2.15.7).

Conclusion

The nuclear design bases for the reference plant are applicable to the TPC designs. Because TPBARs function like conventional burnable absorbers, the core physics of the TPC designs will be comparable to typical reference plant core designs. The following sections describe the first and equilibrium cycle TPC designs and characterize their performance in terms of typical power distributions, reactivity feedbacks, and shutdown margins.

2.4.3.4 Core Design Descriptions

The following gives a description of the first and equilibrium cycle TPC designs. In addition to describing the designs, this section provides information on the tritium production as well.

First Cycle TPC Design

Table 2.4.3-3 gives the fuel region description for the first cycle design. All of the fuel in this design is fresh. A total of 3342 TPBARs and 12,928 IFBA pins were used. Standard Westinghouse IFBA patterns of 128, 80, and 64 IFBA pins per assembly were used. The IFBA pins employ a ^{10}B coating of 1.5 mg/in. This is a standard (designated 1x) coating thickness. The reference plant has routinely employed 1.5x and 1.25x IFBA coating thicknesses. The thinner coating was chosen for the TPC designs in order to make the IFBA fuel rod design more benign with respect to rod internal pressure limits and pellet/clad gap reopening. A secondary effect of the thinner IFBA coating is to very slightly increase BOL fuel temperatures (due to the slightly larger pellet-clad gap). These temperatures have been incorporated into the evaluations of Section 2.15.

As mentioned in the introduction, enrichment zoning was used in the TPC designs to shape the intra-assembly power distribution, reduce power peaking, and slightly increase the tritium production. The zoning pattern used is given in Figure 2.4.3-1. As Table 2.4.3-3 indicates, enrichment zoning was used in all but the low enrichment region (2.80 w/o region). Four

region average enrichments were employed--2.80, 4.091, 4.499, and 4.795--designated LOW, MED1, MED2, and HIGH in the table.

Figure 2.4.3-2 gives the core loading pattern (eighth-core symmetric) for the first cycle. In the RCCA locations, LOW and MED1 fuel regions were used. All other locations have TPBAR clusters and employ MED1, MED2, or HIGH fuel. The highest enrichment fuel is placed on the core periphery. The RCCA locations are shown in Figure 2.4.3-3, which also provides the full core assembly grid.

The TPBARs employed in this design have an active length of 127.5 inches (cold) and are shifted downward from the core midplane by 0.5 inches to shape the axial power distribution. The IFBA pins in regions MED1, MED2, and HIGH have the same axial orientation as the TPBARs. In the LOW region, however, the IFBAs are 135.75 inches long and are shifted downward from the core midplane by 0.25 inches. The axial orientation of the TPBARs and IFBAs specified in this design (and in the equilibrium cycle) should be considered representative. The specific orientation used here is not necessarily optimum from a design margin or tritium production point of view. Like the TPBAR ^{6}Li loading, the axial orientation should be considered a design variable, one of the tools at the core designer's disposal to find the right balance between cycle energy goals, design margins, and tritium production.

Two secondary source clusters will be placed in full core locations H-03 and H-13. These clusters will each have four fresh secondary source rodlets. The source clusters will be designed to be mechanically compatible with the TPBAR clusters of 20 TPBARs slated for those core locations. For symmetry, clusters of 20 TPBARs will be used in the symmetric assembly locations, C-08 and N-08. Primary source rods will be placed in core locations G-02 and G-14. The TPBAR clusters in these locations will have 23 TPBARs instead of 24.

Table 2.4.3-4 gives the core depletion summary including best estimate values for the critical boron concentration and steady state power peaking factors as a function of burnup.

Equilibrium Cycle Design Description

Table 2.4.3-5 gives the fuel region description for the equilibrium cycle design. In this design, 53 once-burned fuel assemblies are used in RCCA locations, while the remainder of the core, 140 locations, contains feed assemblies. A total of 3344 TPBARs and 14,176 IFBA pins are used. As in the first core, the IFBA pins employ a ^{10}B coating of 1.5 mg/in. As Table 2.4.3-5 indicates, the enrichment zoning of Figure 2.4.3-1 was used in all regions. Two region average enrichments were used, 4.304 and 4.744, designated L0X and H0X in the table. The once-burned regions for the low and high enrichment fuel are designated L1X and H1X, respectively.

Figure 2.4.3-4 gives the loading pattern (quarter-core symmetric) for the equilibrium cycle. The TPBARs employed in this design have an active length of 128.5 inches (cold) and are shifted downward from the core midplane by 0.25 inches to shape the axial power distribution. The IFBA pins in all the regions have the same axial orientation as the TPBARs.

As in the first cycle, two secondary source clusters will be placed in full core locations H-03 and H-13. These clusters will have been activated in the first cycle. Primary source rods will not be needed after the first cycle since the secondary source rods will be activated.

Table 2.4.3-6 gives the core depletion summary for the equilibrium cycle design including best estimate values for the critical boron concentration and steady state power peaking factors as a function of burnup.

Nuclear Design Parameter Comparisons

Figure 2.4.3-5 shows a comparison between the hot full power (HFP) critical boron letdown curves for the TPC designs and a recent reference plant core design. Similarly, Figures 2.4.3-6, -7, and -8 show comparisons for $F_{\Delta H}^{N}$, F_{Q}^{N}, and axial offset. All values are best estimate.

Table 2.4.3-8 provides a comparison of various nuclear design parameters (best estimate), including miscellaneous reactivity coefficients, kinetics parameters, control rod worths, fluxes, and boron concentrations.

Tritium Production

Table 2.4.3-7 summarizes the tritium production for the first cycle and equilibrium cycle. Note that the first cycle core produces about 2860 grams of tritium, while the equilibrium cycle produces about 2805 grams. Thus, the average production of tritium per TPBAR is 0.856 gm for the first cycle and 0.839 gm for the equilibrium cycle. The maximum tritium production in any TPBAR is approximately 1.047 grams, well below the 1.2 gram limit. The equilibrium cycle produces slightly less tritium than the first cycle due to the lower average thermal flux that results from the higher (on average) feed enrichments.

Figures 2.4.3-9 and -10 give the first cycle and equilibrium cycle tritium production (grams per TPBAR) for each core location. Two numbers are given for each assembly: the average grams per TPBAR for the entire TPBAR cluster, and the average grams per TPBAR in the maximum channel of the assembly (four channels per assembly).

Design Variations

The designs presented here should be considered representative. As discussed in the introduction, the primary design goal was to employ the maximum number of TPBARs to generate a large amount of tritium. Design variations which generate less tritium but have improved fuel economy are clearly feasible. While not done here, use of TPBARs in once-burned fuel is a loading pattern option which could allow for better fuel economy. Other fuel design options, such as natural or enriched axial blankets with solid or annular pellets, are not specifically precluded by TPBARs, but may require slightly different ^{6}Li loadings or axial orientation. Core designs which employ these fuel design features can and would be analyzed with the same standard design methods used here.

2.4.3.5 Power Distributions

In this section, typical radial and axial power distributions for the TPC designs will be presented. Reference 1, Section 4.3.2.2, provides a general discussion of radial and axial power distributions. This discussion applies to the TPC designs as well. The standard methods used to generate and analyze power distributions for the reference plant (References 4 and 16) have been used here for the TPC designs.

One of the primary constraints in loading pattern development is the nuclear enthalpy rise hot channel factor, $F_{\Delta H}^{N}$. The HFP limit on $F_{\Delta H}^{N}$ assumed in the reference safety analysis is 1.65.
[

][a,c] The TPC designs were analyzed using the methods of Reference 4 to ensure that the $F_{\Delta H}^{N}$ limit was met. Figures 2.4.3-11 through 2.4.3-16 show various radial power distributions for the TPC first cycle. Figures 2.4.3-11, -12, -13, and -14 are HFP, ARO, radial power distributions at the following respective conditions: BOL no xenon, BOL equilibrium xenon, MOL, and EOL. Figures 2.4.3-15 and -16 give HFP BOL and Near EOL power distributions with control bank D to its HFP insertion limit. The RCCA insertion limits used in this study are identical to those routinely used by the reference plant. Also provided in these figures are the peak rod power in each assembly, the assembly burnup distribution, and the lithium-6 fraction remaining for those assemblies containing TPBARs. Figures 2.4.3-17 through -22 give the corresponding figures for the equilibrium cycle.

In terms of power peaking, these power distributions are comparable to those of the reference plant. Just as for the reference plant core designs, there is a slight increase in $F_{\Delta H}^{N}$ in the early part of the cycle (see Figure 2.4.3-6), followed by a decrease with burnup. The TPC designs have slightly lower power peaking in the latter part of the cycle due in part to the reactivity hold down of the TPBARs. The best estimate $F_{\Delta H}^{N}$ design limit is met with significant margin. The powers in the peripheral assemblies for these designs are slightly larger than for the low leakage loading patterns typically used in the reference plant.

Figures 2.4.3-23 and -24 provide BOL and EOL pin power distributions for a typical assembly containing TPBARs. The assembly chosen is from the TPC first cycle. In Figure 2.4.3-2, its location is row 4, column 4. Note in Figures 2.4.3-23 and 2.4.3-24 that the fuel pin powers next to the TPBARs are depressed relative to the peak pin power in the assembly.

Figures 2.4.3-25 and -26 give HFP, normal operation, steady axial power distributions (P(z)) at BOL and EOL, respectively. Power shapes for the TPC designs as well as for a recent cycle of the reference plant are provided. At BOL, the TPC power shapes are somewhat flatter than the shape for the reference due to the part-length TPBARs. As mentioned in the previous section, the active lengths and lithium-6 linear loading of the TPBARs are considered design variables. The lengths used here are likely the shortest that would be used for cores with unblanketed fuel. Cores with natural or slightly enriched axial blankets may benefit from slightly shorter TPBAR active lengths. Longer active lengths than those used here, (e.g., a few inches longer) would cause slightly higher, but very likely acceptable, axial power peaking in the central region of the core. Also, longer lengths would increase tritium production and require higher feed enrichments, unless the linear loading of lithium-6 is reduced. Higher feed enrichments

result since more lithium-6 is being added to the core. Also, the ends of the TPBARs experience the least depletion. Thus, lengthening the TPBARs will increase their residual reactivity penalty. The choice of an active length depends, therefore, on the specifics of the core being designed, requiring a balance between the design margins, cycle energy goals, fuel management constraints (e.g., enrichment limits), and tritium production goals.

Figure 2.4.3-26 shows that the TPC EOL power shapes are similar to the characteristic double-humped shape typically observed.

The RAOC methodology of Reference 16 was used to generate Condition I and Condition II power shapes for the TPC designs for comparison to the F_Q^T and overpower kW/ft limits and for use in the clad stress evaluation. In RAOC, power shapes are generated at various power levels and burnups for a wide range of xenon shapes and covering the allowed operating spaces for control rod insertion and axial flux difference (AFD). As permitted by the Technical Specifications (Reference 1), the AFD envelope is specified each operating cycle as part of the Core Operating Limits Report (COLR). The allowable AFD envelope assumed in these analyses is given in Figure 2.4.3-27. For the TPC first cycle, a slightly narrower AFD envelope was used on the positive side early in the cycle to limit power peaking in the top of the core. Also the First Cycle employed a narrower envelope on the negative side to increase clad stress margin. Figure 2.4.3-28 shows the maximum $F_Q^T(z)$ values for the TPC designs in relation to the $F_Q^T(z)$ limit assumed in the reference plant safety analysis. [

]^{a,c} As Figure 2.4.3-28 shows, the $F_Q^T(z)$ limit is met with considerable margin.

Overpower protection prevents fuel damage and maintains fuel integrity during overpower transients caused by either operator errors or control rod malfunctions. The exact overpower protection setpoints are described in the reference plant Technical Specifications. To meet the overpower requirements, the linear power density during transients should not exceed the 22.4 kW/ft limit.

Two categories of overpower transients are considered. The first category involves control rod malfunctions as well as operator errors in control rod positioning. Control rod malfunctions include rod bank withdrawal accidents. The second category involves cooldown and accidental boration and dilution accidents. These accidents are assumed to occur during any time in life and during normal operating procedures. The results of the Condition II analyses for the TPC designs show that the linear power does not exceed the 22.4 kW/ft limit during postulated overpower transients.

2.4.3.6 Reactivity Coefficients

Section 4.3.2.3 of Reference 1 provides a discussion of reactivity coefficients which is generally applicable to the TPC designs. With only one exception, which is described below, the reactivity coefficients and kinetics parameters for the TPC designs fall within the bounding ranges normally assumed in the reference plant safety analysis.

Table 2.4.3-8 provides typical reactivity coefficients and kinetics parameters for the TPC designs and a recent reference plant core design. In Table 2.4.3-8, note that the BOL MTC values are slightly more negative for the TPC designs than for the reference plant. This is due, in part, to the slightly lower soluble boron worths (boron coefficients) in the TPC designs. The boron worth is less because the core average total absorption cross section for the TPC designs will be greater due to the large inventory of burnable absorbers and higher enrichments. This reduces the thermal neutron flux and the soluble boron reaction rate, resulting in a lower boron worth. Also, for the first cycle, the critical boron concentrations are generally lower than for the reference plant. A lower critical boron concentration will cause the MTC to be more negative. The EOL MTCs for the TPC designs are comparable to those for the reference plant. Figure 2.4.3-29 shows the HFP MTC values as a function of burnup. The boron coefficients in Table 2.4.3-8 reflect the increased thermal neutron absorption mentioned above and the lower boron concentration in the first cycle, i.e., they are less negative than for the reference plant core design at both BOL and EOL.

The Doppler-only power coefficients and Doppler temperature coefficients for the TPC designs are generally comparable to the values for the reference core, but are slightly less negative at BOL. This is likely due to fuel management differences. The core average burnups for the TPC designs are smaller than for typical reference plant core designs because of the large number of feed assemblies employed. As a consequence, there is less ^{240}Pu in the TPCs, which will tend to make the Doppler feedback slightly less negative. The Doppler coefficients in Table 2.4.3-8 do not include any axial flux redistribution effects.

As Table 2.4.3-8 shows, the total power coefficients are always negative at all power levels and throughout life. The power coefficients given here include the Doppler and moderator feedback effects as well as axial flux redistribution effects. At BOL, the power coefficients are more negative for the TPC cores, reflecting the more negative MTCs. At EOL, they are slightly more positive, reflecting less axial flux redistribution.

Figures 2.4.3-30 and -31 give the Doppler-only power defects at BOL and EOL, respectively, for the TPC designs and the reference plant. Similarly, Figures 2.4.3-32 and -33 give the total power defect. Note that the TPC designs exhibit slightly smaller defects than the reference plant core design, reflecting the reactivity coefficient behavior described above.

The delayed neutron fraction and prompt neutron lifetime values given in Table 2.4.3-8 reflect the large feed assembly inventories of the TPC designs. The β_{eff} values are larger than for the reference plant due to the lower inventory of ^{239}Pu. The ℓ^* values are smaller for the TPC designs, reflecting the larger total absorption. The core average flux values given in Table 2.4.3-8 show that the fast and >1 Mev fluxes are comparable to those of the reference plant. The thermal fluxes, however, are slightly smaller.

Table 2.4.3-9 provides the reactivity and kinetics parameter assumptions used in the reference plant safety analysis. The TPC designs fall within the limits and ranges assumed with the exception of the least negative Doppler-only power defect. The TPC designs have BOL defects which are slightly less negative than the safety analysis assumption of -0.998 %$\Delta\rho$ (see Figure 2.4.3-30). A bounding value of -0.92 %$\Delta\rho$ was employed for the TPC designs in the evaluations of Section 2.15.

2.4.3.7 Control Rod Worths and Shutdown Margin

Individual control and shutdown bank worths are highly loading pattern dependent. Table 2.4.3-8 gives individual control bank worths at BOL and EOL for the TPC designs. Values for the reference plant are provided as well. The total worths for all the shutdown RCCAs are also shown. The reference plant loading pattern chosen for comparison here employed fresh fuel in the RCCA locations (see Figure 2.4.3-3). This is evident, for example, in the shutdown RCCA worth at EOL, which is larger than for the TPC designs.

The total rod worth less the most reactive stuck rod is the most important rod worth for determining the core shutdown margin. Table 2.4.3-10 gives EOL shutdown margin assessments for the TPC designs and the reference plant. EOL is limiting for shutdown margin since the power defect is much larger at EOL due the very negative moderator coefficient. The discussion of the individual components of the control rod requirements provided in Reference 1, Section 4.3.2.4, is applicable to the TPC designs and will not be repeated here. As Table 2.4.3-10 shows, the control rod requirements for the TPC designs are comparable to the requirement for the reference plant core design. Similarly, the rod worths (ARI less most reactive stuck rod) are also comparable. The result is that the excess shutdown margins, i.e., the shutdown margin in excess of the 1.30 %$\Delta\rho$ requirement, for the TPC designs are similar to that of the reference plant. The TPC designs meet the 1.30 %$\Delta\rho$ requirement with considerable margin.

2.4.3.8 Effects of Extended Shutdown

When tritium decays (12.33 year half-life), it becomes helium-3. Helium-3 is a potent neutron absorber with a cross section comparable to that of boron-10. Consequently, if a plant has an extended shutdown, there will be a negative reactivity effect due to the buildup of helium-3 if the shutdown occurs at a time when the tritium inventory is significant.

To examine this effect, a calculation was made for the equilibrium cycle TPC in which the core was shut down for 6 months after approximately 80% of the cycle length (17000 MWD/MTU) had elapsed. A 6 month shutdown was chosen as a bounding assumption since long shutdowns near EOL will most likely lead to a decision by the utility to refuel. During this long shutdown, the tritium inventory in the TPBARs was allowed to decay into helium-3. The HZP critical boron concentration was calculated at the end of the shutdown. The core was then returned to full power, and the normal depletion was allowed to continue.

Table 2.4.3-11 shows the critical boron concentrations calculated for this scenario. Two sets of values are given: one in which the tritium is not allowed to decay, and another including the tritium decay and subsequent helium-3 buildup. Note that immediately upon startup after the 6 month shutdown, the effect of the additional helium-3 is a reduction in the critical boron concentration of 80 ppm. The helium-3 will then slowly burn away with full power operation and the critical boron concentration difference will decrease. As with any burnable absorber, however, the burnout is gradual. For this scenario, the helium-3 buildup is enough to shorten the end-of-full-power-capability burnup. At the design EOL of 21564 MWD/MTU, the boron difference is 31 ppm, and the tritium decay case shows a negative boron value of -24 ppm,

indicating that a power coastdown would be needed to achieve that burnup. While only the equilibrium cycle has been examined here, the behavior of the first cycle TPC would be similar.

Sensitive information has been removed

2.4.3.9 Summary

In this section, the two TPC designs have been presented. The design bases employed for these designs are the same as those for conventional cores. In these designs, the TPBARs function in a manner that is similar to standard Pyrex burnable absorbers or WABAs. While the depletion behavior of the TPBARs is different than that of conventional burnable absorbers, this does not lead to significant differences in core physics behavior. The behavior of the designs with respect to power distributions, reactivity coefficients, control rod worths, etc., is quite comparable to typical reference plant core designs. Calculation and analysis of key safety parameters has demonstrated that, with the exception of the least negative Doppler defect, the key safety parameters fall within the ranges and limits normally assumed for the reference plant. As discussed in Section 2.15, this exception does not invalidate the conclusions of the safety analysis.

Based on the above, it is concluded that viable TPC designs can be developed which achieve typical cycle energy goals, generate large amounts of tritium, and meet typical design and safety limits. These designs would be specific to the applicant plant, as would any reload design.

Discussion of the fuel management of the TPC designs and its relationship to fuel storage requirements is given in Section 2.9.2.

2.4.3.10 References

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Table 2.4.3-1
Core Design and Operating Parameters and Selected Design Limits

Parameter	TPC	Reference Core
Number of fuel assemblies	193	193
Number of control rods (RCCAs)	53	53
Control rod material	Ag-In-Cd (80/15/5)	Ag-In-Cd (80/15/5)
Core power level (MW_{th})	3565	3565
Average linear power density (kW/ft)	5.68	5.68
System pressure (psia)	2250	2250
HZP moderator temperature (°F)	557.0	557.0
HFP core average moderator temperature (°F)	589.7	589.7
Fuel Lattice and Assembly Design	17 x 17 VANTAGE+	17 x 17 VANTAGE+
Fuel Rod OD (in. cold)	0.360	0.360
Fuel Pellet OD (in. cold)	0.3088	0.3088
Cladding and Guide Thimble Material	ZIRLO™	ZIRLO™
TPBAR ^{6}Li Linear Loading (gm/in)	0.030	N/A
IFBA ^{10}B Linear Loading (gm/in)	0.0015	0.00188
Active Fuel Height (in. cold)	144	144
Target Cycle Length (MWD/MTU)	21,564	20,600
Target Effective Full Power Days	494	476
Core Loading (MTU)	81.6	82.3
Design $F_{\Delta H}^{N}$ Limit	1.65 (including uncertainties)	1.65 (including uncertainties)
Design F_{Q}^{T} Limit	2.50 (including uncertainties)	2.50 (including uncertainties)
Core Control Strategy	Relaxed Axial Offset Control (RAOC)	Relaxed Axial Offset Control (RAOC)
MTC Limit (pcm/°F)	+7.0 up to 70% power, +0 at 100% power	+7.0 up to 70% power, +0 at 100% power
Shutdown Margin Requirement (%$\Delta\rho$)	1.30	1.30
TPBAR Tritium Production Limit (gm)	1.20	N/A
Fuel Enrichment Limit (w/o U235)	5.0	5.0

Table 2.4.3-2 Definition of Symbols, Terms, and Abbreviations	
Symbol, Term, or Abbreviation	**Definition**
a/o	atom percent
ARI	all rods in
ARO	all rods out
axial offset	$(P_T-P_B)/(P_T+P_B)$. Where P_T is the integrated power in the top half of the core and P_B is the integrated power in the bottom half of the core
β	delayed neutron fraction in core
β_{eff}	effective delayed neutron fraction in core
BA	burnable absorber
BOL	beginning of cycle life
C_B	chemical shim boron concentration in main coolant
CL	centerline of fuel stack
CZP	cold zero power
EOL	end of cycle life
Eq Xe	equilibrium xenon condition
HFP	hot full power
HZP	hot zero power
ΔI	flux difference between the top and bottom halves of the core
IFBA	integral fuel burnable absorber
K(z)	F_Q^T normalized to the maximum value allowed at any core height
ℓ^*	prompt neutron lifetime

Table 2.4.3-2 (Cont.) Definition of Symbols, Terms, and Abbreviations	
Symbol, Term, or Abbreviation	**Definition**
MOL	middle of life
MTC	moderator temperature coefficient
MWD/MTU	megawatt days per metric tonne of initial uranium metal, which represents the integrated energy output of the fuel and is a measure of fuel depletion
pcm	percent mille (reactivity change of 1 pcm equals a reactivity change of $10^{-5}\Delta\rho$)
ppm	parts per million by weight, which specifies chemical shim boron concentration in terms of natural elemental boron by weight
RAOC	Relaxed Axial Offset Control
RCCA	rod cluster control assembly; which is the type of the control rods used in the reference plant
RTP	Rated Thermal Power (3565 MW_{th})
ρ	reactivity
$\Delta\rho$	change in reactivity, defined as $\ln(k_2/k_1)$ where k_1 and k_2 are eigenvalues obtained from two calculations that differ only in the values assigned to the independent variables
shutdown margin (SDM)	the amount of negative reactivity by which a reactor is maintained in a subcritical state
step	a unit of control rod travel equal to 0.625 inches
TPBAR	Tritium Producing Burnable Absorber Rod
T_{mod}	moderator temperature, defined as the temperature corresponding to the average water enthalpy in the core
T_{eff}	resonance effective temperature of fuel
VAMT	Vessel Average Moderator Temperature
WABA	Wet Annular Burnable Absorber
w/o	weight percent

Table 2.4.3-2 (Cont.)
Definition of Symbols, Terms, and Abbreviations

Notes to Table:

Power distributions are quantified in terms of hot channel factors. These factors are a measure of the peak pellet power within the reactor core and the total energy produced in a coolant channel and are expressed in terms of quantities related to the nuclear or thermal design namely:

Power density is the thermal power produced per unit volume of the core (kW/liter).

Linear power density is the thermal power produced per unit length of active fuel (kW/ft). Since fuel assembly geometry is standardized, this is the unit of power density most commonly used. For all practical purposes it differs from kW/liter by a constant factor which includes geometry effects and the fraction of the total thermal power which is generated in the fuel rods.

Average linear power density is the total thermal power produced in the fuel rods divided by the total active fuel length of all rods in the core.

Local heat flux is the heat flux at the surface of the cladding (BTU/hr-ft^2). For nominal rod parameters this differs from linear power density by a constant factor.

Rod power or rod integral power is the length integrated linear power density in one rod (kW).

Average rod power is the total thermal power produced in the fuel rods divided by the number of fuel rods (assuming all rods have equal length).

The hot channel factors are defined as follows:

F_Q^T, the **Heat Flux Hot Channel Factor**, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux for all rods in the core, allowing for manufacturing tolerance on fuel pellets and rods.

F_Q^N, the **Nuclear Heat Flux Hot Channel Factor**, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod parameters.

F_Q^E, the **Engineering Heat Flux Hot Channel Factor**, is the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod, and eccentricity of the gap between pellet and clad. Combined statistically, the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^N$, the **Nuclear Enthalpy Rise Hot Channel Factor**, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It is convenient for the purposes of discussion to define subfactors of F_Q^T. However, design limits are set in terms of the total peaking factor.

Table 2.4.3-2 (Cont.)
Definition of Symbols, Terms, and Abbreviations

Notes to Table: (continued)

F_Q^T = Total peaking factor or heat flux hot channel factor without densification effects.

F_Q^T = (Maximum kW/ft) / (Average kW/ft)
= $F_Q^N \times F_Q^E$
= $\max(F_{xy}^N(z) \times P(z)) \times F_U^N \times F_Q^E$
where: F_Q^N and F_Q^E are defined above

F_U^N = a factor for measurement conservatism, assumed to be 1.05.

$F_{xy}^N(z)$ = the ratio of peak power density to average power density in the horizontal plane at elevation z.

P(z) = the ratio of the power per unit core height in the horizontal plane at elevation z to the average value of power per unit core height.

Table 2.4.3-3 TPC First Core Fuel Region Description							
Region Identifier	**Fuel Type**	**Number of Assemblies**	**Initial U-235 Average Enrichment***	**High Enrich./# of Rods per Assembly**	**Low Enrich./# of Rods per Assembly**	**Number of TPBARs in Region**	**Number of IFBAs in Region**
LOW	VANTAGE+	33	2.800	N/A	N/A	0	2112
MED1	VANTAGE+	52	4.091	4.40 / 196	3.20 / 68	768	2496
MED2	VANTAGE+	60	4.499	4.95 / 196	3.20 / 68	1422	7680
HIGH	VANTAGE+	48	4.795	4.95 / 196	4.35 / 68	1152	640

*Note: Intra-assembly enrichment zoning was employed for all but the low enrichment region.

Table 2.4.3-4
TPC First Cycle
Depletion Summary
(all values are best estimate)

Cycle Burnup (MWD/MTU)	Critical Boron (ppm)	F_Q^N	$F_{\Delta H}^N$	F_Z^N	Axial Offset (%)
0	1458	1.565	1.424	1.058	-1.01
150	1076	1.577	1.397	1.092	-5.00
1000	1113	1.613	1.386	1.134	-6.17
2000	1187	1.665	1.401	1.175	-6.98
3000	1228	1.716	1.421	1.192	-6.95
5000	1220	1.694	1.435	1.169	-4.24
7000	1129	1.610	1.425	1.123	-2.45
9000	997	1.573	1.410	1.099	-1.96
11000	853	1.568	1.395	1.100	-2.53
13000	700	1.546	1.382	1.100	-2.60
15000	539	1.531	1.367	1.099	-2.54
17000	374	1.512	1.350	1.103	-2.63
19000	209	1.493	1.333	1.107	-2.89
21000	47	1.471	1.317	1.107	-3.15
21564	2	1.476	1.313	1.112	-3.42

Table 2.4.3-5 TPC Equilibrium Cycle Fuel Region Description							
Region Identifier*	**Fuel Type**	**Number of Assemblies**	**Initial U-235 Average Enrichment***	**High Enrich./# of Rods per Assembly**	**Low Enrich./# of Rods per Assembly**	**Number of TPBARs in Region**	**Number of IFBAs in Region**
H1X	VANTAGE+	49	4.744	4.95 / 196	4.15 / 68	0	2848
L1X	VANTAGE+	4	4.304	4.60 / 196	3.45 / 68	0	0
H0X	VANTAGE+	112	4.744	4.95 / 196	4.15 / 68	2672	9792
L0X	VANTAGE+	28	4.304	4.60 / 196	3.45 / 68	672	1536

*Note: "1X" refers to once-burned fuel and "0X" refers to fresh fuel

*Note: Intra-assembly enrichment zoning was employed for all regions.

Table 2.4.3-6 TPC Equilibrium Cycle Depletion Summary (all values are best estimate)					
Cycle Burnup (MWD/MTU)	Critical Boron (ppm)	F_Q^N	$F_{\Delta H}^N$	F_z^N	Axial Offset (%)
0	1752	1.569	1.436	1.064	2.18
150	1341	1.600	1.441	1.064	-2.56
1000	1326	1.589	1.426	1.087	-3.24
2000	1337	1.599	1.446	1.107	-3.58
3000	1335	1.639	1.453	1.126	-3.90
5000	1274	1.632	1.443	1.126	-3.85
7000	1163	1.586	1.419	1.114	-3.41
9000	1019	1.534	1.386	1.102	-3.24
11000	868	1.508	1.361	1.104	-3.51
13000	709	1.480	1.333	1.101	-3.41
15000	547	1.455	1.314	1.098	-3.19
17000	381	1.430	1.296	1.097	-3.03
19000	216	1.409	1.277	1.099	-2.99
21000	53	1.398	1.260	1.102	-3.20
21564	8	1.396	1.258	1.102	-3.28

Table 2.4.3-7
Tritium Production for the
First and Equilibrium Cycle Conceptual Core Designs

Parameter	First Cycle	Equilibrium Cycle
Number of TPBARs	3342	3344
Initial ^{6}Li Linear Loading (gm/in)	0.030	0.030
Absorber Height (in)	127.5	128.5
Average ^{6}Li Fraction Remaining	0.546	0.558
Average Grams of Tritium Produced Per TPBAR	0.856	0.839
Peak Grams of Tritium Produced Per TPBAR*	~1.047	~1.044
Total Grams of Tritium Produced	2860	2805

* Average over 6 TPBARs in the assembly channel producing the most tritium (4 channels per assembly). The true maximum may be slightly higher.

Table 2.4.3-8 Nuclear Design Parameters (all values best estimate)			
Parameter Description	**Reference Plant Recent Cycle**	**TPC First Cycle**	**TPC Eq. Cycle**
Reactivity Coefficients			
Moderator Temperature Coefficients (pcm/°F)			
Near BOL, HZP, No Xenon	4.7	2.7	1.3
BOL, HFP, Eq. Xenon	-7.3	-9.7	-9.9
EOL, HFP, Eq. Xenon	-32.8	-31.7	-32.9
Boron Coefficients (pcm/ppm)			
BOL, HZP	-7.6	-7.2	-6.3
BOL, HFP	-7.3	-6.9	-6.0
EOL, HZP	-9.8	-8.4	-7.6
EOL, HFP	-9.5	-8.1	-7.5
Doppler-Only Power Coefficients (pcm/% Power)			
BOL, HZP	-13.2	-11.4	-11.2
BOL, HFP	-7.9	-7.7	-7.5
EOL, HZP	-10.8	-10.5	-10.5
EOL, HFP	-7.6	-7.6	-7.5
Total Power Coefficients (pcm/% Power)			
BOL, HZP	-13.4	-16.7	-15.7
BOL, HFP	-10.4	-10.9	-10.9
EOL, HZP	-32.2	-28.8	-29.8
EOL, HFP	-26.6	-23.7	-24.7
Doppler Temperature Coefficients (pcm/°F)			
BOL, HZP	-1.8	-1.6	-1.7
BOL, HFP	-1.3	-1.2	-1.3
EOL, HZP	-1.9	-1.9	-1.9
EOL, HFP	-1.5	-1.5	-1.5

Table 2.4.3-8 (Continued)
Nuclear Design Parameters
(all values best estimate)

Parameter Description	Reference Plant Recent Cycle	TPC First Cycle	TPC Eq. Cycle
Kinetics Parameters			
Delayed Neutron Fraction, β_{eff}			
BOL	0.00619	0.00695	0.00653
EOL	0.00505	0.00532	0.00532
Prompt Neutron Lifetime (ℓ*), μ sec			
BOL	14.11	13.69	11.80
EOL	17.23	14.69	13.40
HZP Control Rod Worths (pcm)			
Bank D BOL / EOL*	557 / 655	503 / 601	555 / 591
Bank C BOL / EOL	1045 / 1235	1048 / 1261	1148 / 1147
Bank B BOL / EOL	747 / 1127	980 / 1053	860 / 851
Bank A BOL / EOL	794 / 645	593 / 705	645 / 660
Shutdown Banks BOL / EOL	2771 / 4283	3083 / 3790	3559 / 3497
*BOL with No Xenon, EOL with HFP Eq. Xenon			
HFP Core Average Neutron Fluxes (n/cm^2-sec)			
BOL			
Thermal	4.40E13	4.26E13	3.67E13
Fast	3.17E14	3.14E14	3.17E14
> 1 MeV	8.6E13	8.4E13	8.5E13
EOL			
Thermal	5.39E13	4.65E13	4.23E13
Fast	3.25E14	3.28E14	3.28E14
> 1 MeV	8.8E13	8.8E13	8.8E13
Thermal Flux < 0.65 eV, Fast Flux > 0.65 eV			

Table 2.4.3-8 (Continued) Nuclear Design Parameters (all values best estimate)			
Parameter Description	**Reference Plant Recent Cycle**	**TPC First Cycle**	**TPC Eq. Cycle**
<u>Boron Concentrations (ppm)</u>			
HFP, ARO, BOL, No Xenon, Critical	1650	1458	1752
HFP, ARO, BOL, Eq. Xenon, Critical	1305	1076	1341
HZP, ARO, BOL, No Xenon, Critical	1803	1625	1942
HZP, ARI, BOL, No Xenon, $k_{eff} = 0.99$	1153	870	1003
CZP, ARI, BOL, No Xenon, $k_{eff} = 0.95$	1762	1829	1979

Table 2.4.3-9 Reactivity Coefficients and Kinetics Parameters Values and Ranges Assumed in Reference Plant Transient Analyses	
Parameter	**Value or Range**
Maximum MTC (pcm/°F)	+7.0 (RTP ≤ 70%) ramping to 0.0 at 100% RTP
Most Positive Moderator Density Coefficient ($\Delta\rho$/gm/cc)	0.50
Doppler Temperature Coefficient (pcm/°F)	-2.9 to -0.91
Doppler-Only Power Coefficient, Zero to Full Power, (pcm/% power)	-19.4 to -6.05
Delayed Neutron Fraction, β_{eff}	0.0046 to 0.0073
Least Negative Doppler-Only Power Defect (%$\Delta\rho$)	
BOL	-0.998
EOL	-0.950

Note: The TPC designs fall within above limits and ranges, with the exception of the least negative Doppler-only power defect at BOL. A bounding value of -0.92 %$\Delta\rho$ was chosen for this parameter for the evaluations of Section 2.15.

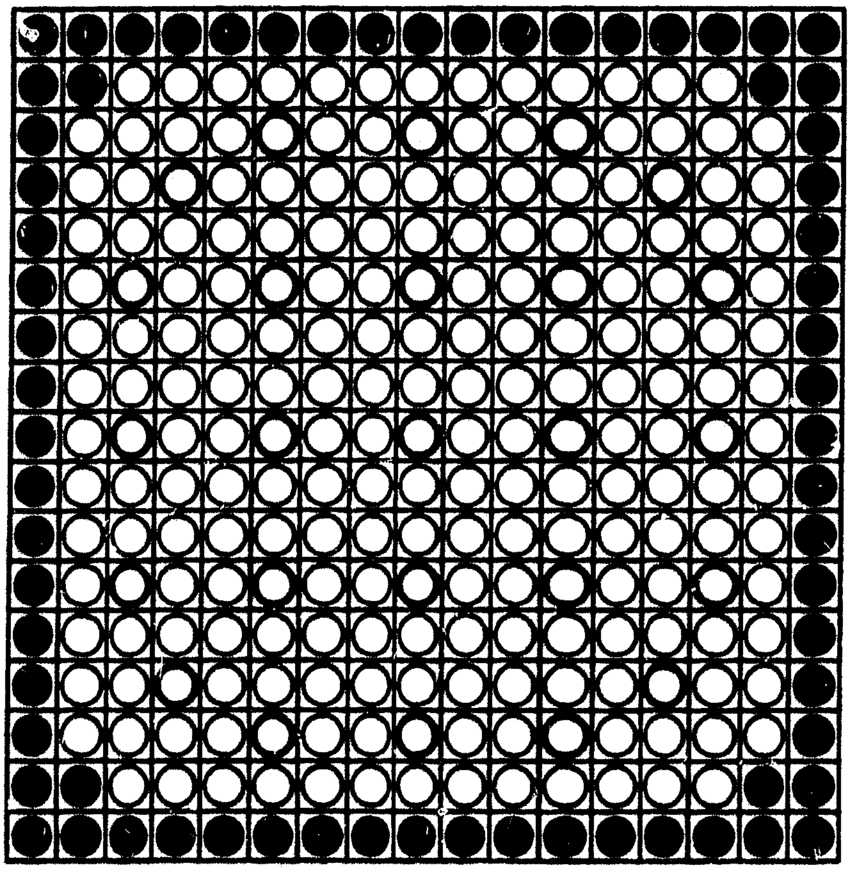
Table 2.4.3-10 EOL Shutdown Requirements and Margins			
SDM Component	**Reference Plant (%Δρ)**	**TPC First Cycle (%Δρ)**	**TPC Eq. Cycle (%Δρ)**
Positive Reactivity Additions			
(1) Total Power Defect*	3.13	2.86	2.95
Negative Reactivity Additions**			
ARI Less Most Reactive Stuck Rod	5.79	5.93	5.43
(2) Less 10%	5.21	5.34	4.89
Shutdown Margin			
(3) Calculated Margin {(2) - (1)}	2.08	2.48	1.94
(4) Required Shutdown Margin	1.30	1.30	1.30
Excess Shutdown Margin {(3) - (4)}	0.78	1.18	0.64

*The total power defect includes the Doppler defect, the moderator defect, axial flux redistribution effects, T_{mod} uncertainties, and void collapse.

**The rod worth value implicitly accounts for the Rod Insertion Allowance.

Table 2.4.3-11
Effect of Helium-3 Build-up on Critical Boron Concentration
After a 6 Month Shutdown for the TPC Equilibrium Cycle

Condition	Critical Boron Assuming No Tritium Decay (ppm)	Critical Boron With Tritium Decay (ppm)	Difference (ppm)
HFP at 17000 MWD/MTU	382	382	0
HZP at shutdown	673	673	0
HZP after 6 month shutdown, no xenon	982	911	71
HFP after 6 month shutdown, no xenon	705	625	80
HFP at 19000 MWD/MTU, eq. xenon	208	153	55
HFP at 21000 MWD/MTU	51	16	35
HFP at 21564 MWD/MTU	7	-24	31



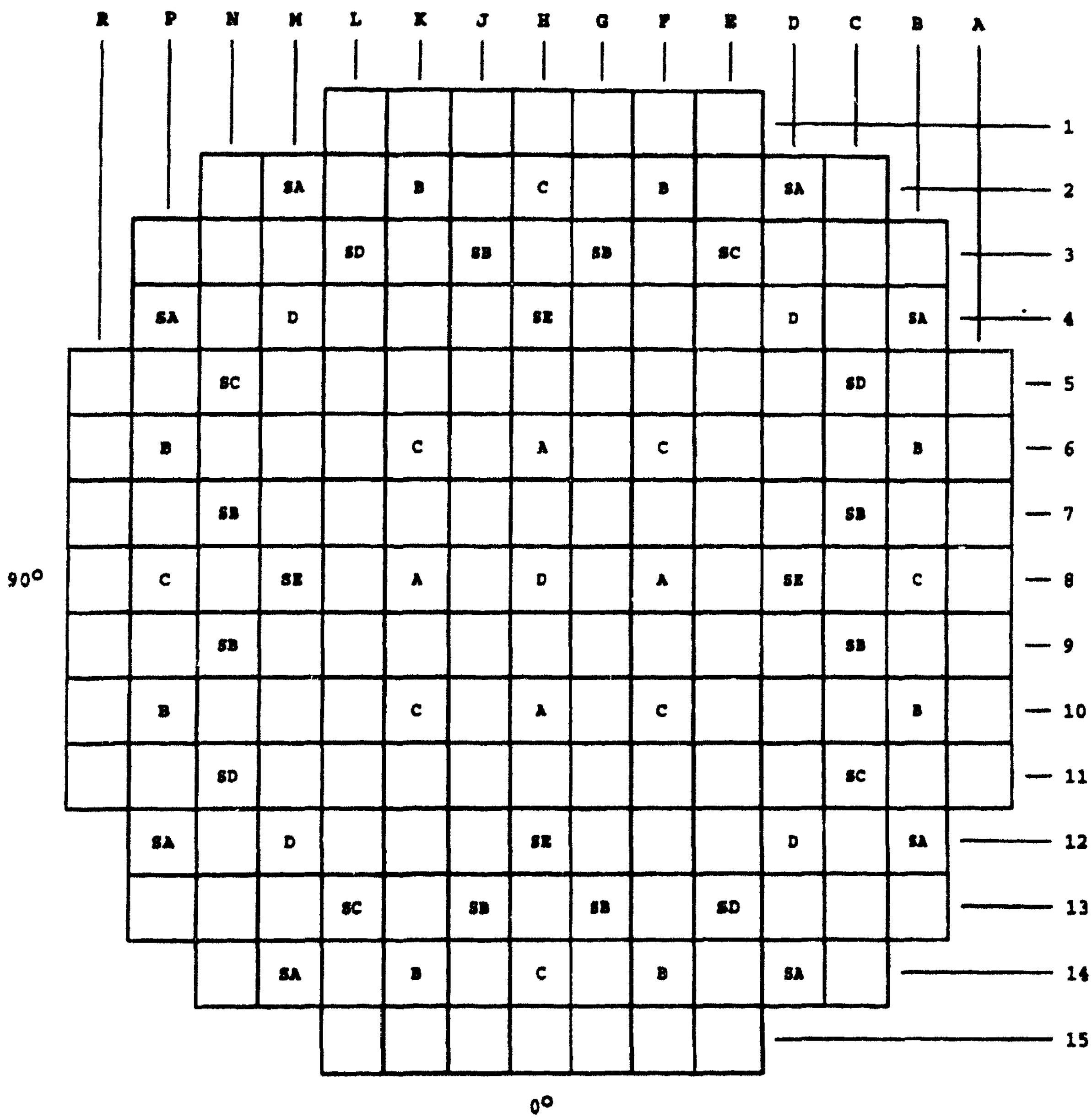
Low Enrichment Fuel Pins (68 rods)
High Enrichment Fuel Pins (196 rods)
Guide Thimble / Instrumentation Thimble

Figure 2.4.3-1
Enrichment Zoning for Tritium Production Cores

	1	2	3	4	5	6
1	LOW 64					
2	MED2 128 24	MED1 80 24				
3	LOW 64	MED2 128 24	LOW 64			
4	MED2 128 24	MED1 24	MED2 128 24	MED1 24		
5	LOW 64	MED1 128 24	MED1 24	MED2 128 24	LOW 64	
6	MED2 128 20	LOW 64	MED2 128 24	LOW 64	MED2 128 24	HIGH 24
7	MED1 128	MED2 128 24	MED1 80	HIGH 80 24	MED1	HIGH 24
8	HIGH 24	HIGH 24	HIGH 24	HIGH 24		

REGION # IFBAs # TPBARs

Figure 2.4.3-2
Tritium Production Core
First Cycle Loading Pattern



Bank Identifier	Number of Locations	Bank Identifier	Number of Locations
A	4	SA	8
B	8	SB	8
C	8	SC	4
D	5	SD	4
		SE	4

Figure 2.4.3-3
Tritium Production Core Designs
Control Rod and Shutdown Rod Locations

	1	2	3	4	5	6	7	8
1	H1X 128	H0X 128 24	H1X 104	H0X 128 24	H1X 104	H0X 128 20	H1X 32	H0X 32 24
2	H0X 128 24	L0X 64 24	H0X 128 24	L0X 64 24	H0X 128 24	H1X 104	H0X 104 24	H0X 32 24
3	H1X 104	H0X 128 24	L1X	H0X 128 24	L0X 64 24	H0X 128 24	H1X 32	H0X 32 24
4	H0X 128 24	L0X 64 24	H0X 128 24	L0X 64 24	H0X 128 24	H1X 32	H0X 104 24	H0X 24
5	H1X 104	H0X 128 24	L0X 64 24	H0X 128 24	H1X 104	H0X 104 24	H1X	
6	H0X 128 20	H1X 104	H0X 128 24	H1X 32	H0X 104 24	L0X 24	H0X 24	
7	H1X 32	H0X 104 24	H1X 32	H0X 104 24	H1X	H0X 24		
8	H0X 32 24	H0X 32 24	H0X 32 24	H0X 24				

H = HIGH ENRICH.
L= LOW ENRICH.
0X = FRESH FUEL
1X=ONCE-BURNED

REGION # IFBAs # TPBARs

Figure 2.4.3-4
Tritium Production Core
Equilibrium Cycle Loading Pattern

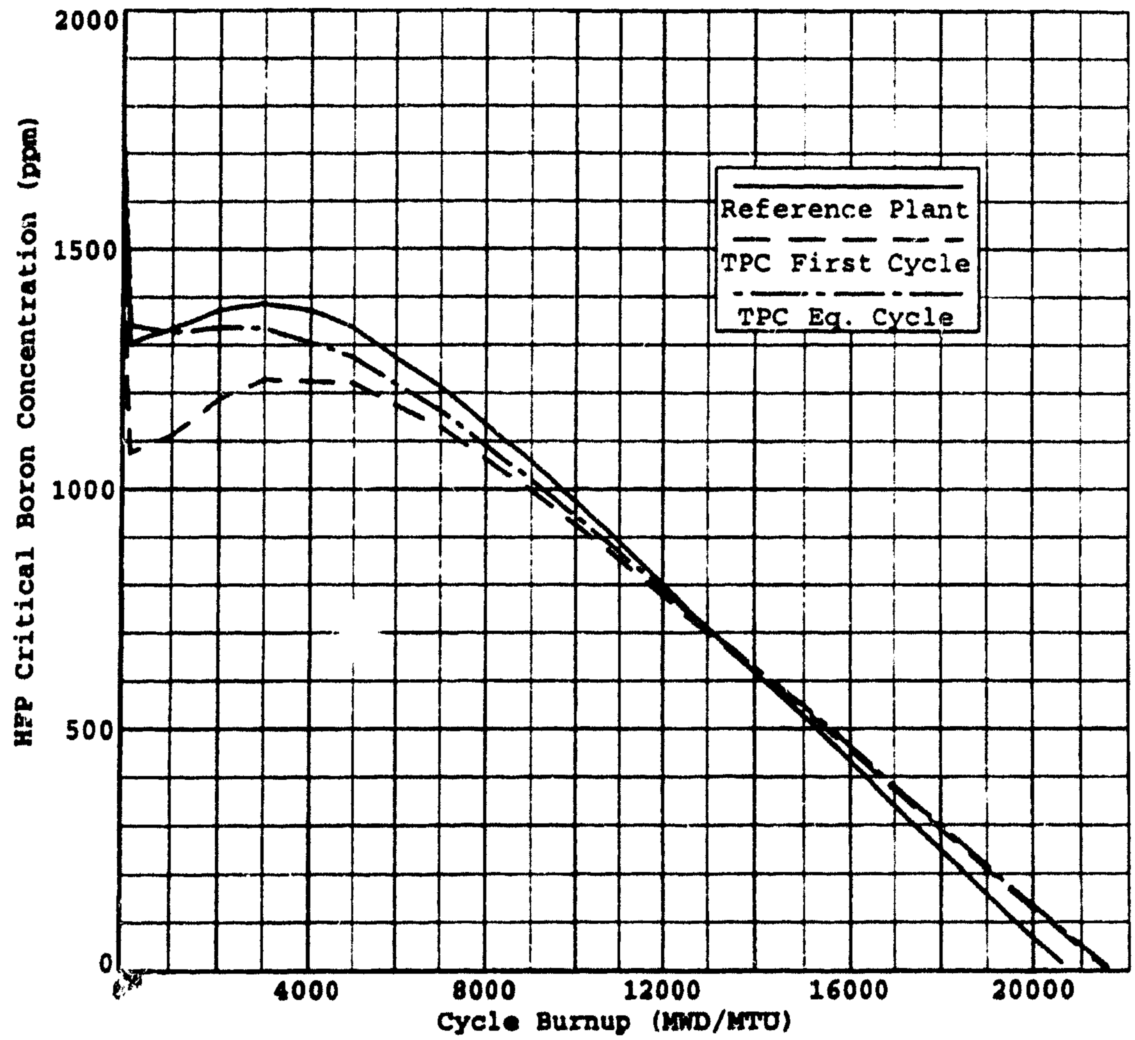


Figure 2.4.3-5
HFP Critical Boron versus Burnup for Reference Plant and TPC Designs

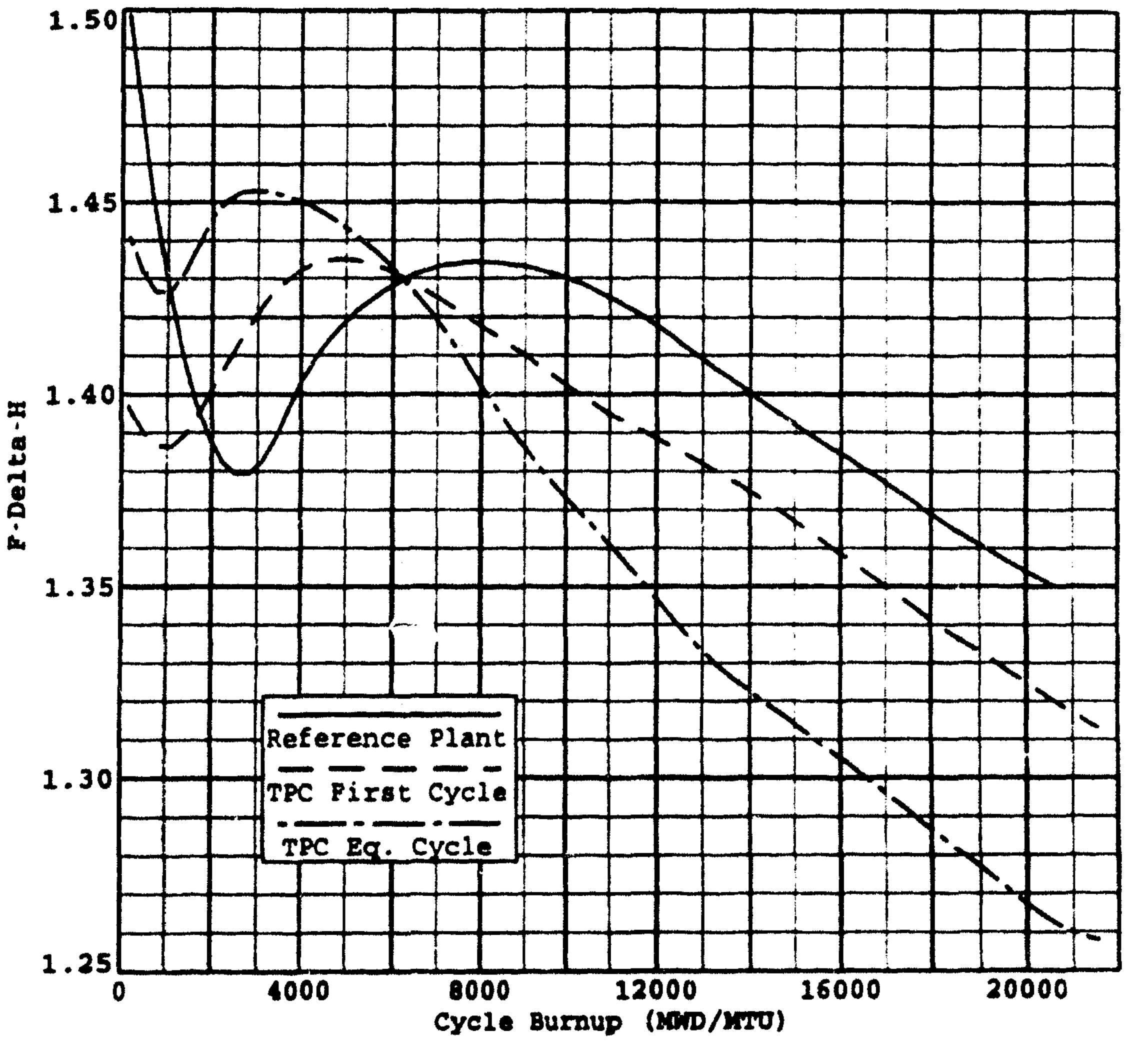


Figure 2.4.3-6
Normal Operation F-Delta-H versus Burnup for Reference Plant and TPC Designs

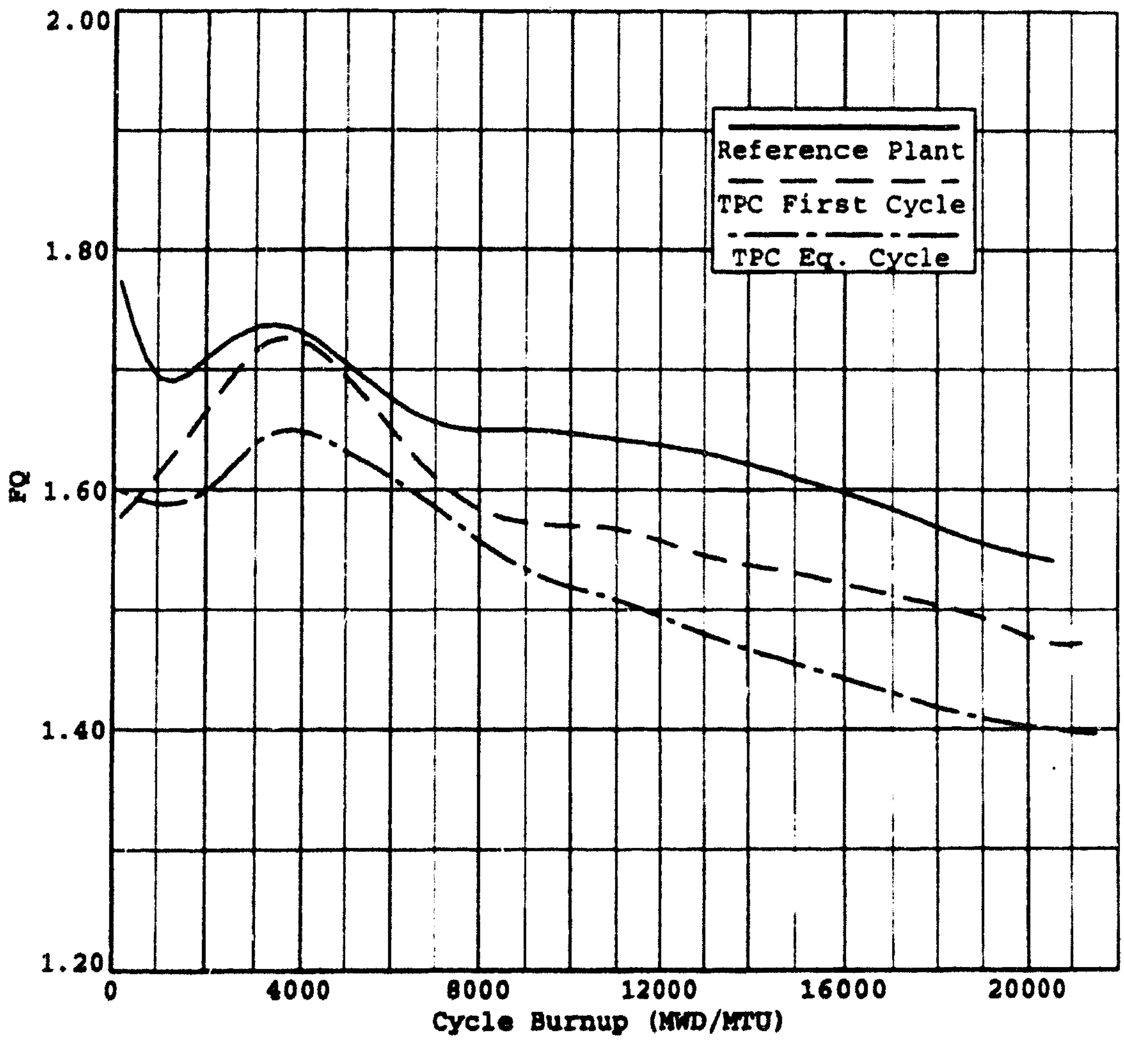


Figure 2.4.3-7
Steady State FQ versus Burnup for Reference Plant and TPC Designs

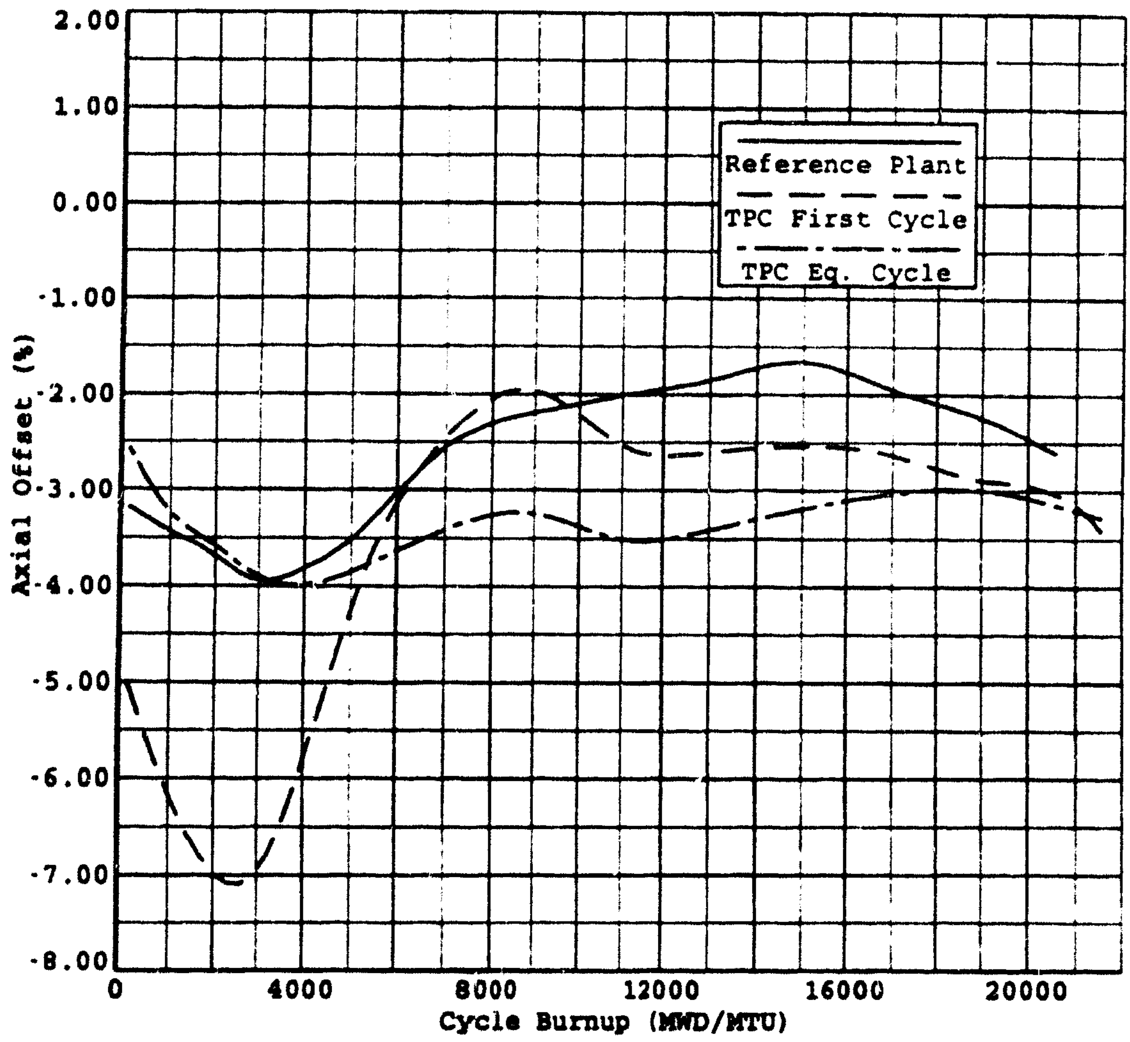


Figure 2.4.3-8
Axial Offset versus Cycle Burnup for Reference Plant and TPC Designs

a,c

Figure 2.4.3-9
Tritium Production Core First Cycle Average and Maximum Tritium Production (grams/TPBAR)

a,c

Figure 2.4.3-10
Tritium Production Core Equilibrium Cycle Average and Maximum Tritium Production (grams/TPBAR)

a,c

REGION IDENT.	NUMBER OF ASSEMBLIES	POWER SHARING	BURNUPS TOTAL	CYCLE
LOW	33	1.065	0	0
MED2	60	1.088	0	0
MED1	52	1.118	0	0
HIGH	48	.718	0	0

Figure 2.4.3-11
Tritium Production Core First Cycle Power and Burnup at 0 MWD/MTU, HFP, No Xenon, ARO

a,c

REGION IDENT.	NUMBER OF ASSEMBLIES	POWER SHARING	BURNUPS TOTAL	CYCLE
LOW	33	1.079	161	161
MED2	60	1.085	163	163
MED1	52	1.113	167	167
HIGH	48	.717	108	108

Figure 2.4.3-12
Tritium Production Core First Cycle Power and Burnup
At 150 MWD/MTU, HFP, Equilibrium Xenon, ARO

a,c

REGION IDENT.	NUMBER OF ASSEMBLIES	POWER SHARING	BURNUPS TOTAL	BURNUPS CYCLE
LOW	33	1.110	12299	12299
MED2	60	1.140	12384	12384
MED1	52	1.087	12093	12093
HIGH	48	.655	7189	7189

Figure 2.4.3-13
Tritium Production Core First Cycle Power and Burnup
At 11000 MWD/MTU, HFP, Equilibrium Xenon, ARO

a,c

REGION IDENT.	NUMBER OF ASSEMBLIES	POWER SHARING	BURNUPS TOTAL	CYCLE
LOW	33	1.071	23814	23814
MED2	60	1.136	24417	24417
MED1	52	1.073	23498	23498
HIGH	48	.703	14351	14351

Figure 2.4.3-14
Tritium Production Core First Cycle Power and Burnup
At 21564 MWD/MTU, HFP, Equilibrium Xenon, ARO

a,c

REGION IDENT.	NUMBER OF ASSEMBLIES	POWER SHARING	BURNUPS TOTAL	CYCLE
LOW	33	1.068	21057	21057
MED2	60	1.136	21502	21502
MED1	52	1.083	20743	20743
HIGH	48	.693	12565	12565

Figure 2.4.3-15
Tritium Production Core First Cycle Power and Burnup
At 150 MWD/MTU, HFP, Equilibrium Xenon, D At 161 Steps

a,c

REGION IDENT.	NUMBER OF ASSEMBLIES	POWER SHARING	BURNUPS TOTAL	BURNUPS CYCLE
LOW	33	1.068	21057	21057
MED2	60	1.136	21502	21502
MED1	52	1.083	20743	20743
HIGH	48	.693	12565	12565

Figure 2.4.3-16
Tritium Production Core First Cycle Power and Burnup
At 19000 MWD/MTU, HFP, Equilibrium Xenon, D At 161 Steps

a,c

REGION IDENT.	NUMBER OF ASSEMBLIES	POWER SHARING	BURNUPS TOTAL	CYCLE
H1X	49	1.148	16031	0
H0X	112	.910	0	0
L0X	28	1.067	0	0
L1X	4	1.240	16540	0

Figure 2.4.3-17
Tritium Production Core Equilibrium Cycle Power and Burnup
At 0 MWD/MTU, HFP, No Xenon, ARO

a,c

REGION IDENT.	NUMBER OF ASSEMBLIES	POWER SHARING	BURNUPS TOTAL	BURNUPS CYCLE
H1X	49	1.153	16204	173
H0X	112	.911	137	137
L0X	28	1.057	159	159
L1X	4	1.227	16725	185

Figure 2.4.3-18
Tritium Production Core Equilibrium Cycle Power and Burnup
At 150 MWD/MTU, HFP, Equilibrium Xenon, ARO

a,c

REGION IDENT.	NUMBER OF ASSEMBLIES	POWER SHARING	BURNUPS TOTAL	BURNUPS CYCLE
H1X	49	1.053	27960	11928
H0X	112	.937	10160	10160
L0X	28	1.130	12323	12323
L1X	4	1.215	30368	13828

Figure 2.4.3-19
Tritium Production Core Equilibrium Cycle Power and Burnup
At 11000 MWD/MTU, HFP, Equilibrium Xenon, ARO

a,c

REGION IDENT.	NUMBER OF ASSEMBLIES	POWER SHARING	BURNUPS TOTAL	BURNUPS CYCLE
H1X	49	1.044	39027	22995
H0X	112	.954	20153	20153
L0X	28	1.091	24057	24057
L1X	4	1.104	42581	26041

Figure 2.4.3-20
Tritium Production Core Equilibrium Cycle Power and Burnup
At 21564 MWD/MTU, HFP, Equilibrium Xenon, ARO

a,c

REGION IDENT.	NUMBER OF ASSEMBLIES	POWER SHARING	BURNUPS TOTAL	BURNUPS CYCLE
H1X	49	1.148	16204	173
H0X	112	.913	137	137
L0X	28	1.055	159	159
L1X	4	1.236	16725	185

Figure 2.4.3-21
Tritium Production Core Equilibrium Cycle Power and Burnup
At 150 MWD/MTU, HFP, Equilibrium Xenon, D At 161 Steps

a,c

REGION IDENT.	NUMBER OF ASSEMBLIES	POWER SHARING	BURNUPS TOTAL	CYCLE
H1X	49	1.039	36347	20316
H0X	112	.953	17711	17711
L0X	28	1.09[illegible]	21249	21249
L1X	4	1.137	39720	23180

Figure 2.4.3-22
Tritium Production Core Equilibrium Cycle Power and Burnup At 19000 MWD/MTU, HFP, Equilibrium Xenon, D At 161 Steps

a,c

Figure 2.4.3-23
BOL First Cycle Pin Power Distribution for Assembly with 24 TPBARs

a,c

Figure 2.4.3-24
EOL First Cycle Pin Power Distribution for Assembly with 24 TPBARs

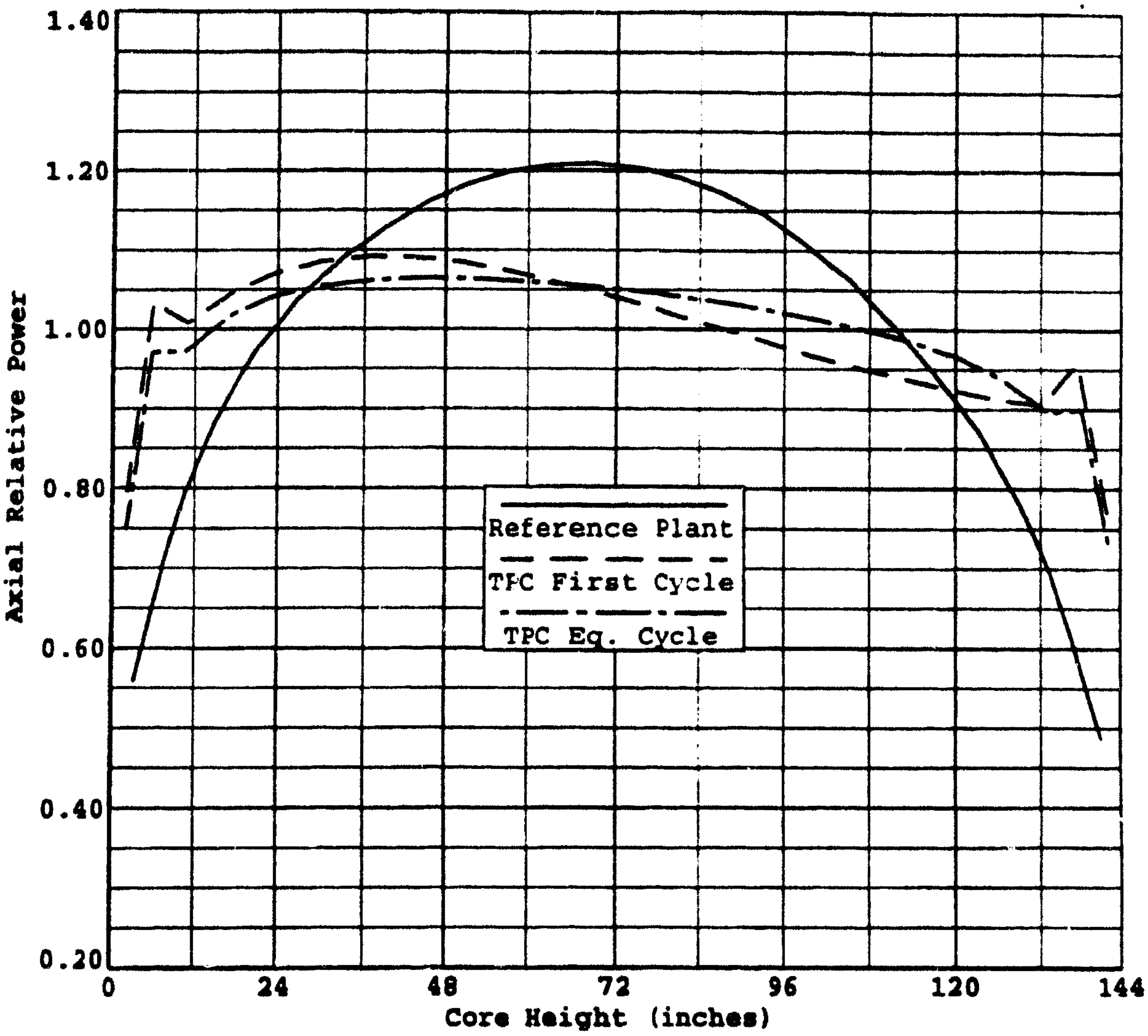


Figure 2.4.3-25
Axial Power Shape Comparison - BOL, HFP for Reference Plant and TPC Designs

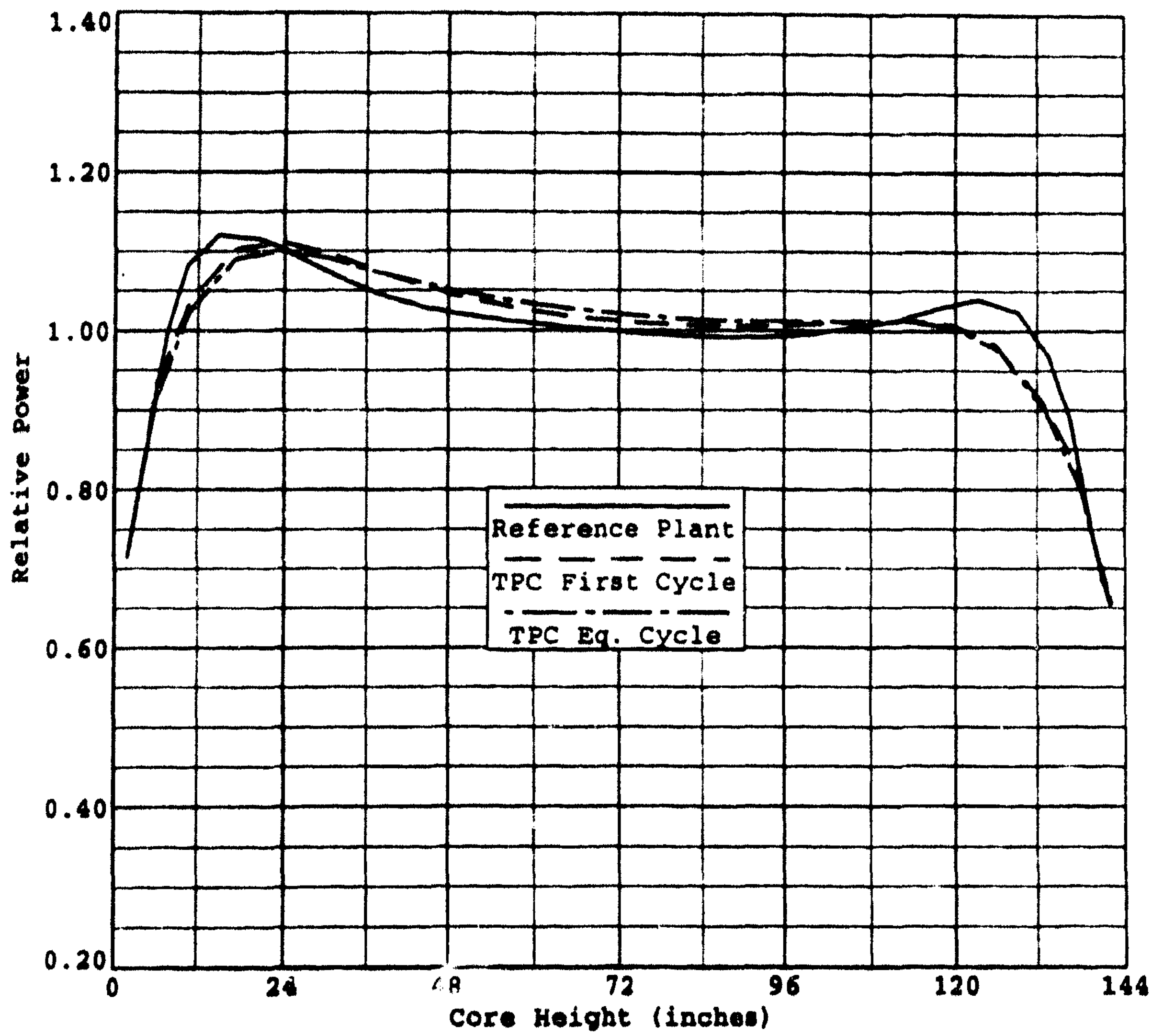


Figure 2.4.3-26
Axial Power Shape Comparison - EOL, HFP for Reference Plant and TPC Designs

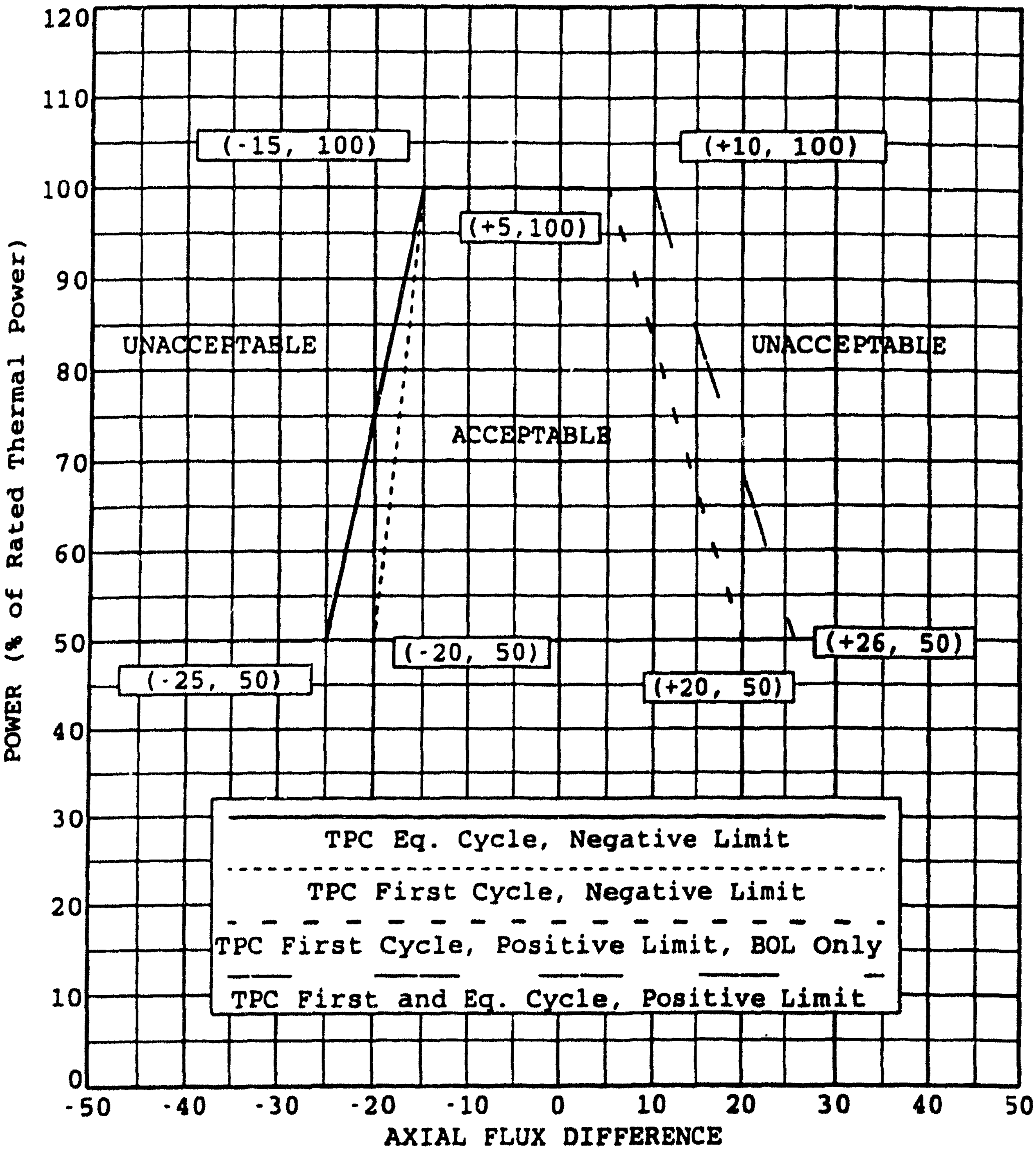


Figure 2.4.3-27
Axial Flux Difference Limits as a Function of Rated Thermal Power

a, c

Maximum $(F_Q^T \times P_{rel})$

Figure 2.4.3-28
Maximum $\left(F_Q^T \times P_{rel}\right)$ versus Core Height During Normal Core Operation

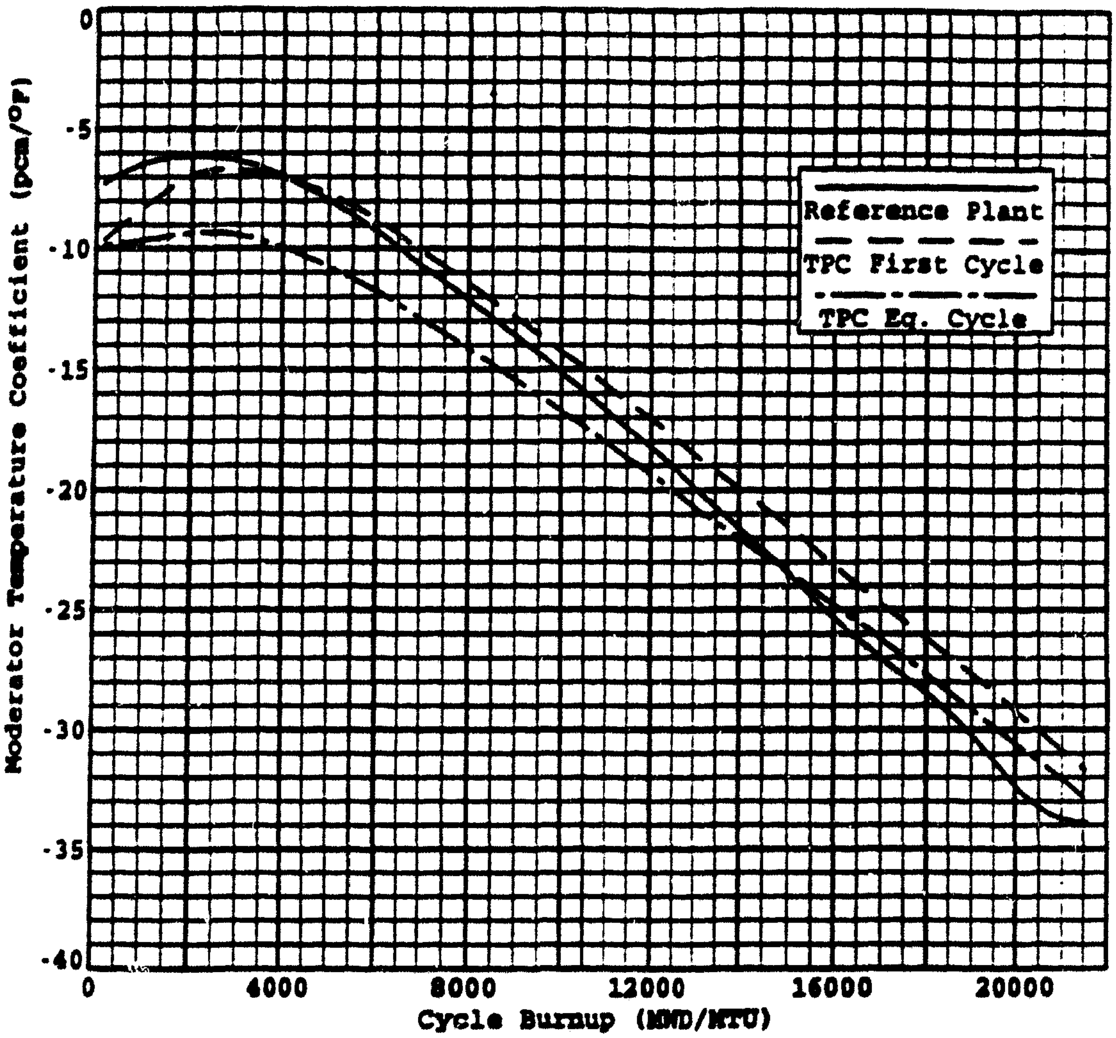


Figure 2.4.3-29
HFP Moderator Temperature Coefficient
For Reference Plant and TPC Designs

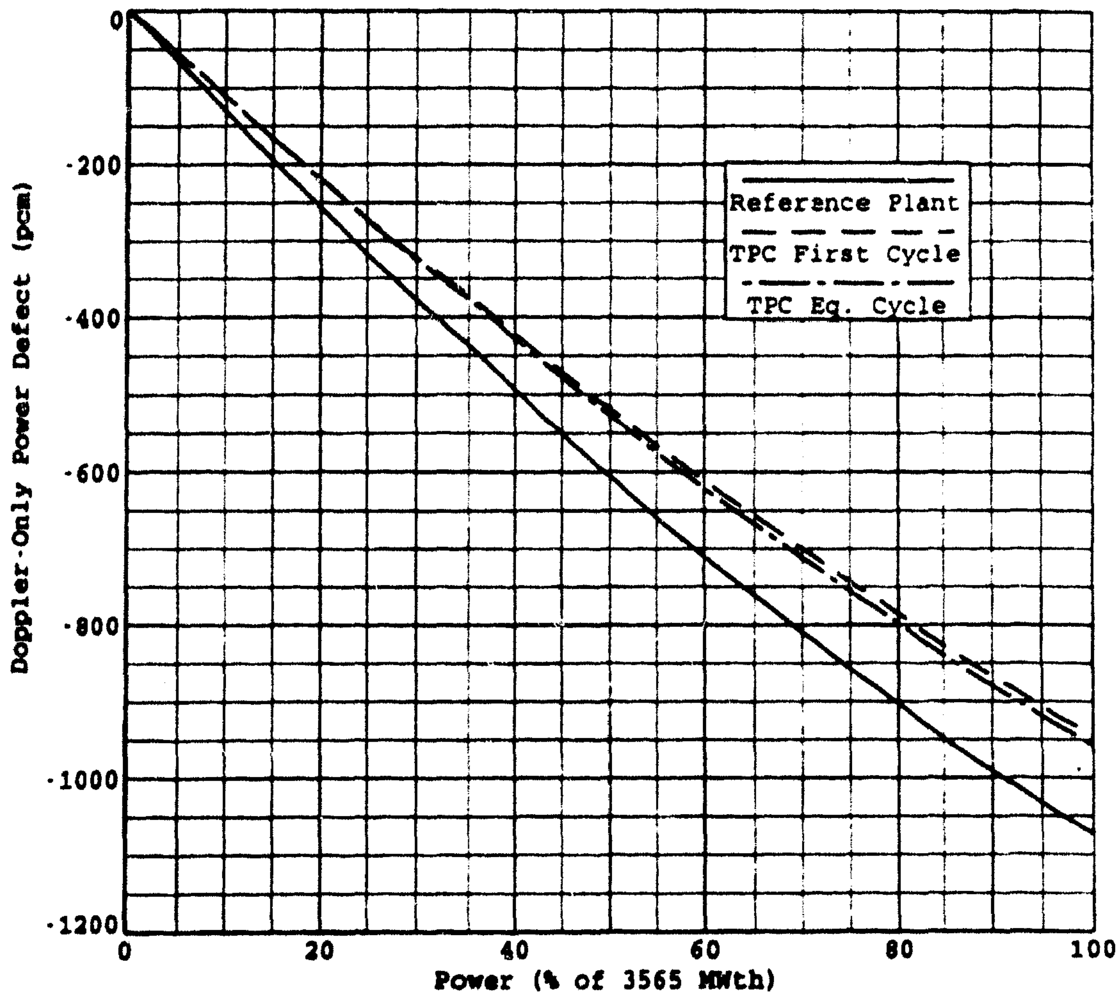


Figure 2.4.3-30
BOL Doppler - Only Power Defect for Reference Plant and TPC Designs

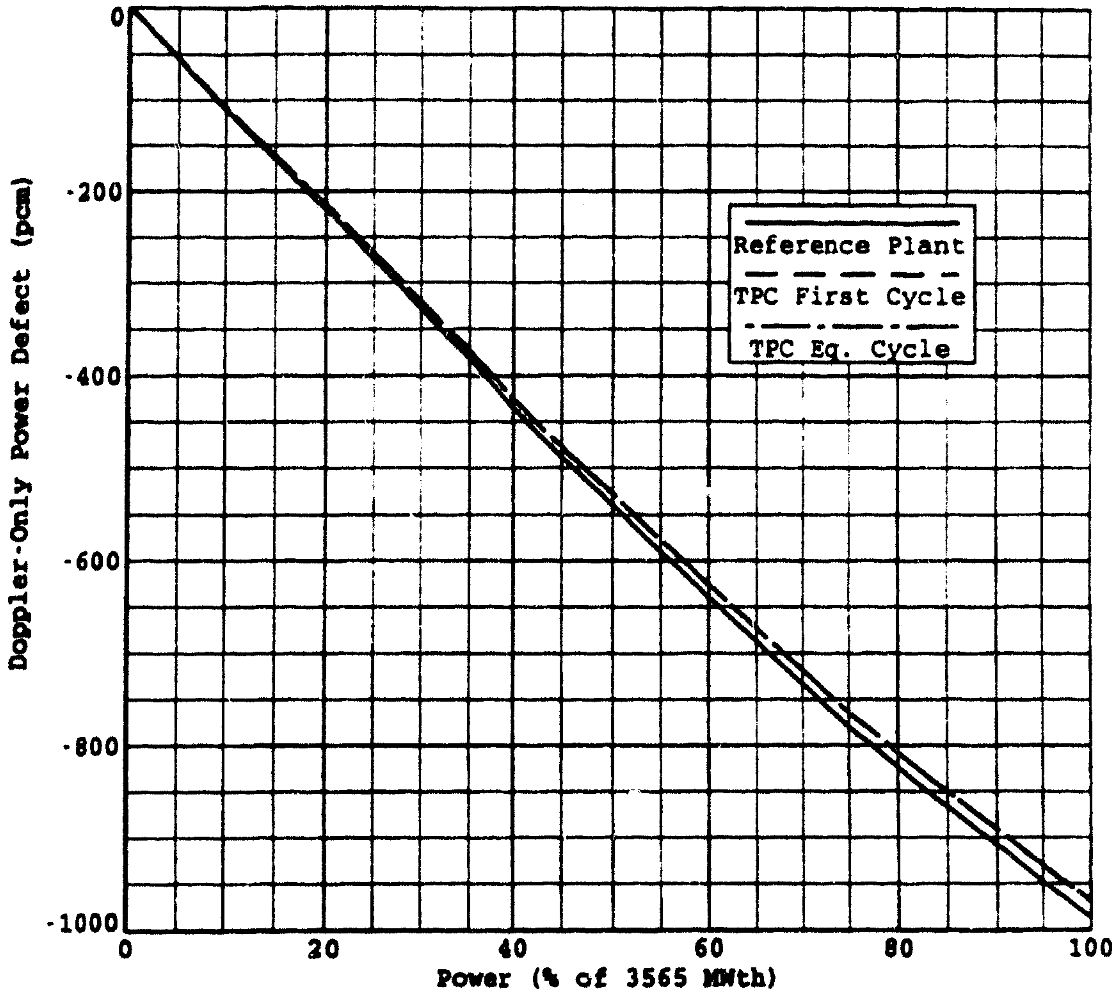


Figure 2.4.3-31
EOL Doppler-Only Power Defect for Reference Plant and TPC Designs

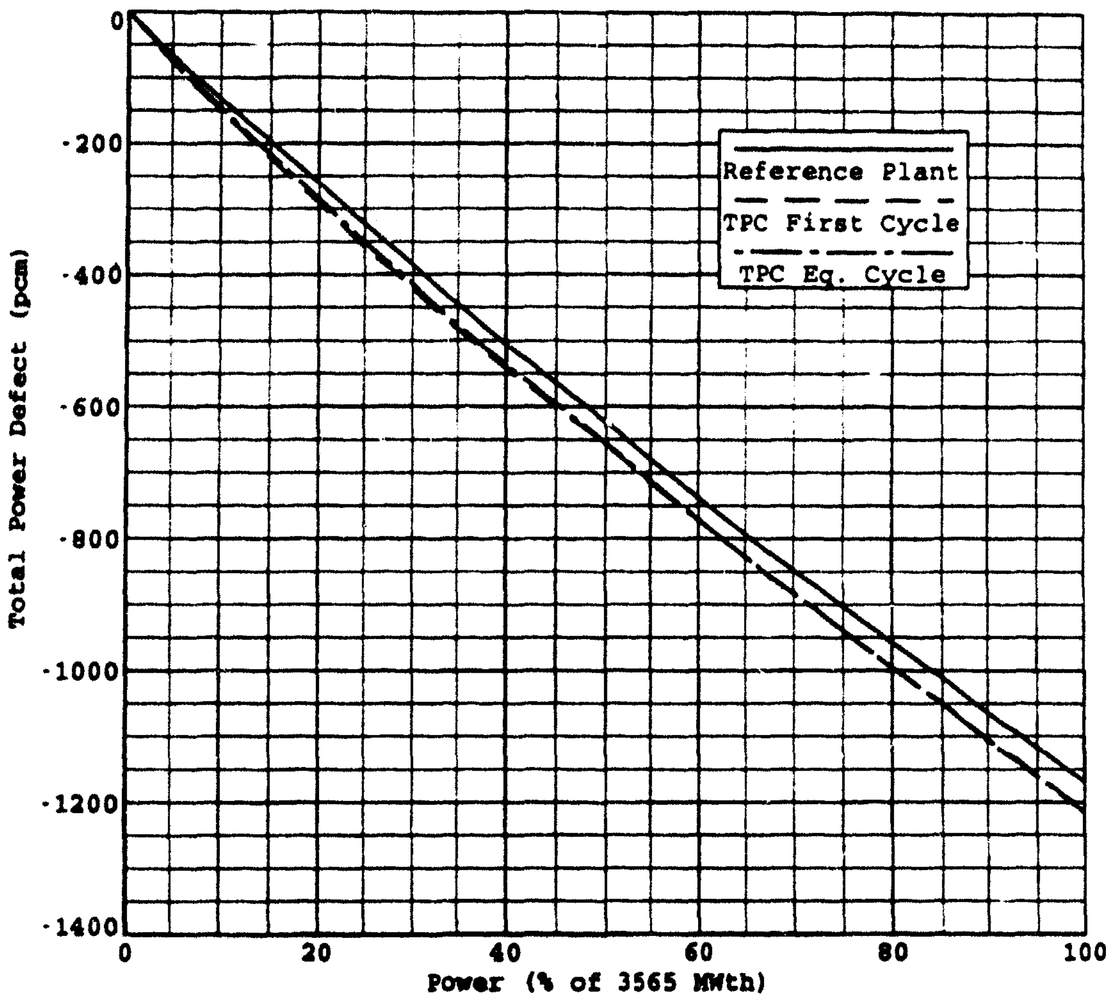


Figure 2.4.3-32
BOL Total Power Defect for Reference Plant and TPC Designs

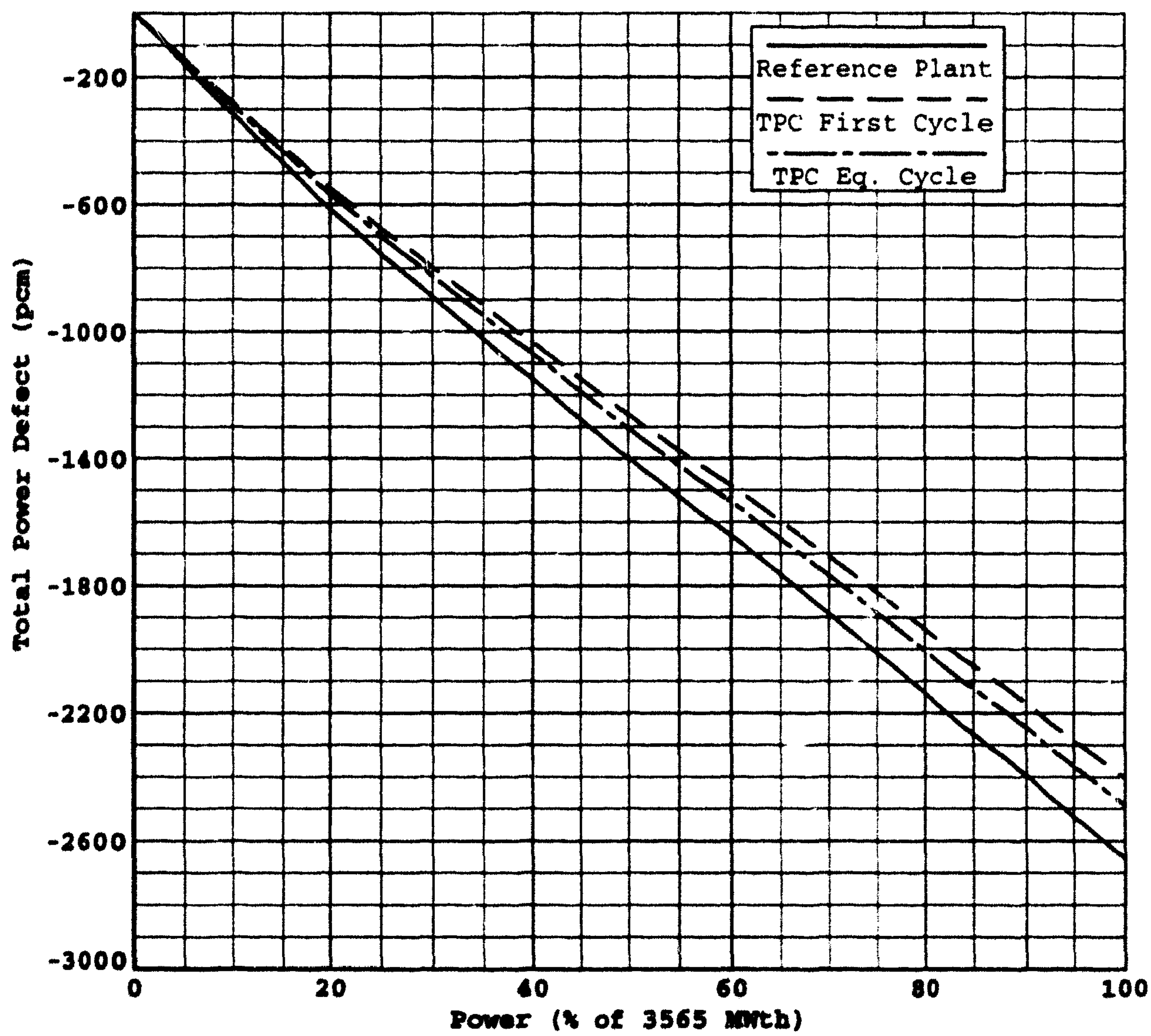


Figure 2.4.3-33
EOL Total Power Defect for Reference Plant and TPC Designs

2.4.4 THERMAL AND HYDRAULIC DESIGN (SRP 4.4) EVALUATION

2.4.4.1 Introduction

Thermal/Hydraulic design of the core is verified to assure safe operation and integrity of the core and fuel assemblies for operation and transients categorized under Condition I and II, up to faults of moderate frequency from which the plant is expected to recover. An evaluation has been performed considering the application of the Tritium-Producing Burnable Absorber Rods (TPBAR) in a reference core for thermal/hydraulic design. The result of this evaluation is a confirmation that the thermal/hydraulic design conditions will continue to be acceptable with respect to design criteria under such a core configuration.

2.4.4.2 Methodology

The methodology that was pursued to arrive at the conclusion of acceptability of TPBARs is consistent with current standard methods for components inserted into cores of Westinghouse design. The presumption was made that the TPBAR would be a core component not unlike other core components that are routinely inserted into the core for purposes of reactivity and power distribution control. Indeed, to avoid mechanical complexity, the outside geometry of the TPBAR was selected so that it was within the envelope of existing burnable absorber component rods.

With insertion of the TPBAR, some performance differences were seen and accommodated, for which input to the evaluation was developed. A calculation was performed with indicated core limits for the reference plant and fuel type with the T/H design code, THINC-IV (References 1 & 2). This result is used to define core conditions, such as core and assembly flow and channel enthalpy rise, which are taken as boundary conditions for the response of the core component.

With the "channel side" conditions in hand from THINC-IV, the thimble, with the core component applied, is then analyzed. The "thimble side"analysis is a calculation of the flow system through the fuel assembly thimble considering the absorber rod's presence. The process takes the core pressure drop and performs an iterative calculation, node-by node, up the rod in an effort to match the pressure drop from the "channel", or open fuel assembly, side to that of the flow through the thimble itself. Consideration of heating rates of the absorber and its internal material characteristics, as well as power peaking in the assembly is included. From this calculation, the absorber temperature, thimble flow, local pressure, component cladding and thimble temperature are found.

Two allied analyses were performed, assuming incorporation of the TPBARs. 1) An axial shape study was performed to assure that more limiting power distributions would not be caused by the presence of the TPBAR component. This compares power shapes resulting from depletion during operation of the cycle with reference shapes used as the basis for thermal/hydraulic design analyses. 2) A Rod Withdrawal from Subcritical transient was analyzed to demonstrate the continued acceptability of the DNBR for this transient. This was done because the least negative Doppler-only power defect with TPBARs is not bounded by the reference plant safety analyses (see Sections 2.4.3.6 and 2.15.2.8.1).

In addition, consideration of a feedwater malfunction event concluded that the effect was minimal and would be bounded by other events.

2.4.4.3 Significant Input Parameters and Assumptions

The geometry of the thimble and core component is taken in a consistently conservative direction with respect to dimensional tolerance for input in the model. The design enthalpy rise peaking factor is assumed for computation of the fuel rod side boundary conditions, as well as for heat generation within the absorber. Generic power distribution profiles are used for thermal/hydraulic design, generally, and all such generic parameters are applied for this TPBAR evaluation.

The model is fairly explicit in terms of loss coefficients for orifice flow paths, friction, etc. Generic, or standard, inputs were used for these values in the calculations performed based on Westinghouse documentation of testing. Projection of core-wide effects, such as on core bypass flow, would be on the assumption that all thimbles were filled with the component as specified, the number being deduced by simple multiplication of calculated results across the number of thimbles in the core.

2.4.4.4 Acceptance Criteria

The objective of the thermal/hydraulic design is to assure the continued integrity of the core. For fuel rods, this is met by demonstrating that DNB is not likely to occur, according to a standard probability and confidence level.

For core components, this is reflected in assuring the mechanical integrity of the thimble and component such that the component remains functional, can be removed/reinserted and that the fuel assembly retains its structural soundness and remains serviceable.

To guard against the debilitating effect of excessive heating and corrosion, the design criterion is taken that the TPBAR component must not exceed its melting temperature. There will be no surface boiling from the core component within the dashpot region of the thimble. There will be no bulk boiling in the thimble along its length, and the sum of the flow through all the thimble/component combinations must be less than allowed by bypass flow limits used to assure adequate flow for core cooling.

2.4.4.5 Results

For flexibility in the potential application of TPBARs, calculations were performed assuming a core with 17x17 VANTAGE+ fuel assemblies with Intermediate Flow Mixing (IFM) grids applied and repeated again assuming no IFMs. The IFMs create mixing turbulence within the flow and improve cooling, which allows greater peaking factors in the core power distribution.

The "thimble side" calculations completed for the TPBAR component (see Section 3.6) have shown that acceptance criteria are met generally; no bulk boiling in the thimble or surface boiling in the dashpot with the application of IFMs. The nominal peaking factor applicable to

the reference plant was 1.65. If the IFMs are not applied, the peaking factor must be limited to a maximum value of [ ]^{a,c} in order to demonstrate compliance to these criteria.

The axial power shape comparison showed that with the assumption of RAOC operation strategy, the power shapes and analysis for the reference plant would remain bounding. The TPBAR would not present any significant power distribution changes beyond those which are already bounded within the thermal/hydraulic design bases.

The Rod Withdrawal from Subcritical analysis was completed, calculating the resultant DNBR for statepoints generated in the safety analysis. The calculated DNBR remained above the Safety Limit DNBR for this event based on the reference plant. Therefore, the acceptability of the TPBAR application was demonstrated.

2.4.4.6 Conclusions & Affected Documents

The work performed and reported here to assess effects of incorporation of TPBARs on the reference plant has demonstrated that these components can be incorporated with little perturbance to the thermal/hydraulic design. Bases will continue to be met for the structural integrity of the assembly due to thermal and hydraulic effects. Bypass flow will remain within its design limit of 8.4%. The DNB criterion will continue to be met with no feature of the TPBAR particularly challenging cooling capacity of the core.

2.4.4.7 Potential Plant-Specific Analytical Issues

There is a sensitivity demonstrated here to flow mixing (with IFM vs. without IFM). This is generally seen, and is not unique to the TPBAR. Of course, for any plant specific application, the current peaking factor / enthalpy rise for the plant must be taken into account. Though the general conclusion of this evaluation is that the TPBAR can be applied without restriction, the continued compliance to criteria given the operating conditions of the particular plant must be individually assured.

2.4.4.8 References

1. WCAP-7956, "THINC-IV - An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," Hochreiter, L. E., Chu, P. T., Chelemer, H., June 1973.

2. WCAP- 8054, "Application of the THINC-IV Program to PWR Design," Hochreiter, L. E. and Chelemer, H., September 1973.

2.4.5 RCCA ROD DROP TIME (SRP 4.6) EVALUATION

2.4.5.1 Introduction

During full power plant operation all Rod Cluster Control Assemblies (RCCAs) and corresponding drive rod assemblies are held at a fully withdrawn position by individual Control Rod Drive Mechanisms (CRDMs). If any operation or accident event necessitates an immediate core power shutdown, all CRDMs will release the RCCAs, allowing them to drop from their fully withdrawn to their fully inserted positions. This gravity induced drop occurs only because total RCCA/drive rod assembly weight is greater than mechanical and hydraulic resistance forces opposing downward RCCA motion.

However, the excess of RCCA/drive rod assembly weight over resistive forces must be large enough that the drop, or SCRAM, times are less than safety criteria specifications for specific operating and external (e.g., earthquake) conditions.

The objective of this section is to summarize the results of the RCCA rod drop time evaluation with and without TPBARs for the reference plant. It is anticipated that the addition of TPBARs will slightly increase the core pressure drop, thus slightly increasing the calculated rod drop time.

2.4.5.2 Methodology

In view of the expenditure involved in testing, a better economic position is realized by calculating actual SCRAM times for feasible driveline design or design alterations, combined with various combinations of plant operating and external conditions, before full scale implementation of the design. In this way, calculated SCRAM times can be compared against criteria to determine whether or not new designs or design alterations would be acceptable for implementation.

The elapsed time for complete, gravity induced, RCCA drop (insertion) into the core is determined using the dynamic forces balance. This force balance:

$$\sum F_y = m\,a$$

can be expanded to address the interactions between the RCCA/drive rod assembly and the remaining components of the driveline and fluid systems.

This expanded form is:

[ ]^b Equation 2.4.5-1

where:

[ ]^{b}

The correlations used to evaluate the individual force terms are dependent upon static fluid pressure differences and mechanical misalignments in the respective driveline regions, and upon correlations made between empirical expressions and test data. Their values are also influenced by geometric and other fluid flow characterizations representative of a given situation. Figure 2.4.5-1 is a simplified RCCA and drive rod assembly representation illustrating the forces affecting RCCA drop.

The fuel assembly thermal/hydraulic parameters used in this analysis were based on results of the analysis discussed in Section 2.4.4. System parameters, such as inlet temperature, reactor coolant pressure and flow were provided in Table 1-1.

The reactor pressure vessel system evaluation was performed based on Westinghouse 17x17 VANTAGE+ fuel with IFM grids.

2.4.5.3 Acceptance Criteria and Results

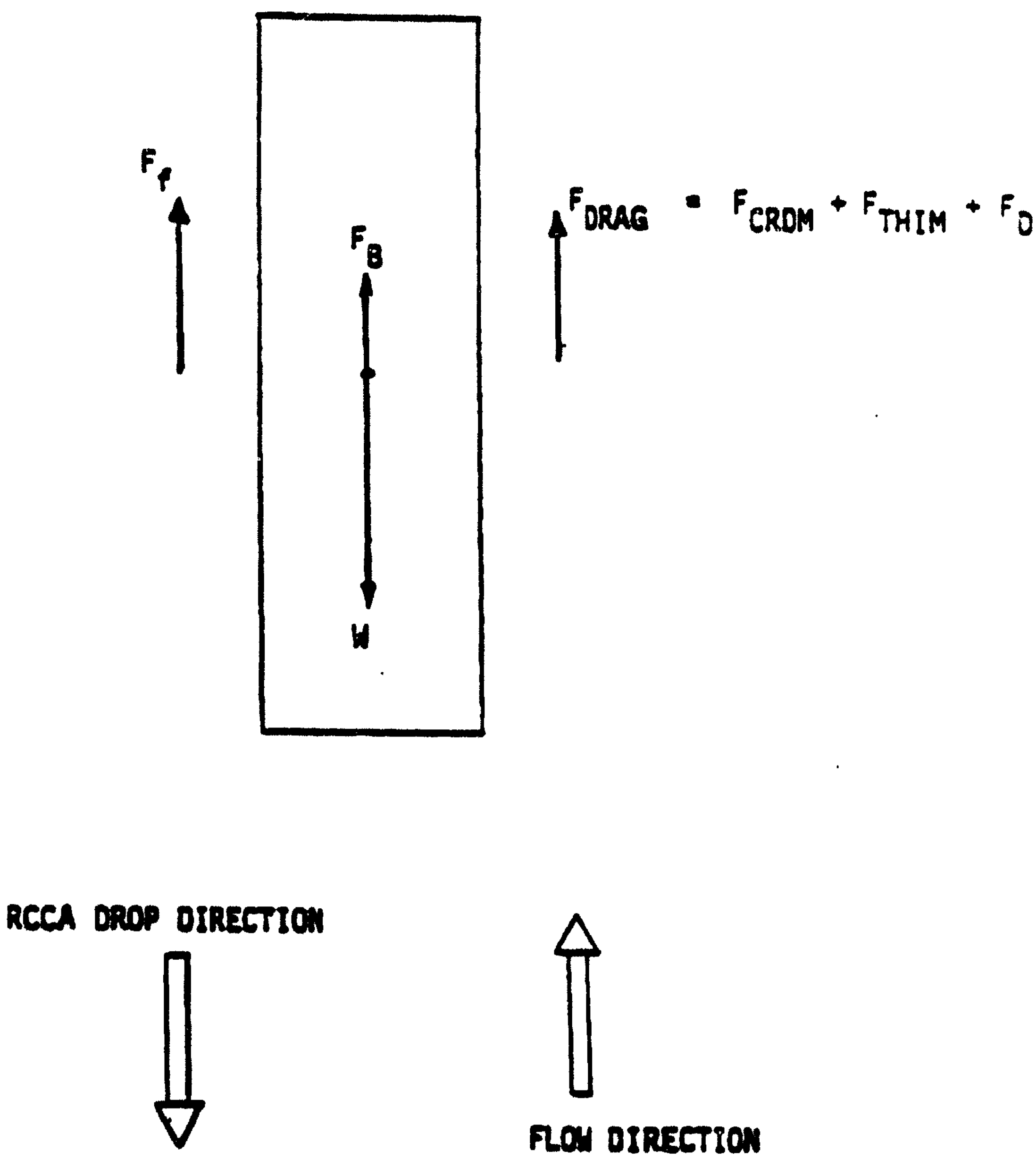
The RCCA rod drop time acceptance criteria is defined by the reference plant Technical Specification limit of 2.7 seconds. This limit represents the time elapsed from initiating the drop to the time it reaches the top of the dashpot. The calculation resulted in a drop time to dashpot entry of [ ]^{b} seconds with TPBARs. The drop time to dashpot entry without TPBARs is [ ]^{b} seconds.

2.4.5.4 Conclusion

Based on the calculations, the RCCA rod drop times are all under the reference plant Technical Specification limit of 2.7 seconds to the top of the dashpot.

Therefore, with TPBARs and at the operating conditions defined in Table 1-1, the RCCA will scram under the Technical Specification time of 2.7 seconds to the top of the dashpot for the reference plant. Overall the effect on RCCA rod drop time of adding the TPBARs is insignificant, increasing the drop time by only [ ][b] seconds.

Figure 2.4.3-12
Tritium Production Core First Cycle Power and Burnup
At 150 MWD/MTU, HFP, Equilibrium Xenon, ARO



Note: See Equation 2.4.5-1

Figure 2.4.5-1
Block Presentation of RCCA and Drive Rod Assembly Illustrating the Relationship Between the Forces Affecting RCCA Drop

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2.5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

2.5.1 INTRODUCTION

The sections in Chapter 5 of the SRP deal with maintaining the integrity of the reactor coolant pressure boundary. Evaluations of the overpressurization protection, reactor vessel materials and pressure-temperature limits, and residual heat removal system are discussed in Sections 2.5.2 through 2.5.4.

SRP 5.2.1.1- 5.2.1.2
Compliance with the Codes and Standards Rule, 10 CFR 50.55a and Applicable Code Cases: The acceptance criteria in these sections are based on the relevant requirements of 10 CFR 50.55a and 10 CFR 50, Appendix A, GDC 1. They specifically deal with the identification of Code Classes for pressure-retaining components of the reactor coolant pressure boundary and other fluid systems important to safety, and the identification of ASME Code Cases which have been applied. The TPBAR pressure boundary (cladding and top and bottom end plugs) is not classified as an ASME Class 1, 2, or 3 component, and therefore this section is not impacted.

SRP 5.2.2
Overpressurization Protection: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 15 and 31. They specifically deal with the adequacy of the relief valves and safety valves in providing overpressure protection at power, and with the adequacy of the low temperature, overpressure protection system (LTOPS) during startup and shutdown below the enable temperature. Criteria specific to LTOPS are based on Branch Technical Position RSB 5-2. The predicted peak pressures during normal operational transients, and during severe abnormal operational transients with reactor scram, are not impacted by the presence of TPBARs (see Sections 2.3.2 and 2.15). An evaluation of the impact of the TPBARs on the LTOPS was performed and is summarized in Section 2.5.2 of this document.

SRP 5.2.3
Reactor Coolant Pressure Boundary Materials: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a; 10 CFR 50, Appendix A, GDC 1, 4, 14, 30 and 31; 10 CFR 50, Appendix B; and 10 CFR 50, Appendix G. They specifically deal with the adequacy and suitability of the ferritic materials, stainless steels, and nonferrous metals used in the reactor coolant pressure boundary (other than the reactor pressure vessel), the compatibility of these materials with the reactor coolant, and the fabrication and processing of these materials. The effects of the TPC on reactor coolant chemistry were evaluated (see Section 3.8.3 for details). In the worst case for normal operation, where two TPBARs are breached, the reactor coolant chemistry remains well within the design limits for impurities and particulates. Given that the reactor coolant chemistry is not significantly affected, and the materials have not changed, there is no impact due to the TPC on this SRP.

SRP 5.2.4
Reactor Coolant Pressure Boundary Inservice Inspection and Testing: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a and

10 CFR 50, Appendix A, GDC 32. They specifically deal with the adequacy of the inservice inspection program including the definition of the system boundary subject to inspection, the accessibility of the components to be inspected, the examination techniques, the inspection interval, the standards for examination evaluation, repair procedures, and the system leakage and hydrostatic pressure test program. The incorporation of the TPBARs does not change the design of the reactor coolant pressure boundary, and therefore does not impact any aspect of the inspection and testing program.

SRP 5.2.5
Reactor Coolant Pressure Boundary Leakage Detection: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2 and 30. They specifically deal with the adequacy of the leakage detection system to identify (to the extent practical) the source of reactor coolant leakage, to separately monitor and collect leakage from both identifiable and unidentifiable sources, to provide indication in the main control room allowing qualitative interpretations for each leakage detection system, and to allow for the monitoring for intersystem leakage. The incorporation of the TPBARs does not cause any new potential leakage paths, therefore, the leakage detection system is not impacted.

SRP 5.3.1
Reactor Vessel Materials: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 55.55a; 10 CFR 50, Appendix A, GDC 1, 4, 14, 30, 31 and 32; 10 CFR 50, Appendix B; 10 CFR 50 Appendix G; and 10 CFR 50 Appendix H. They specifically deal with the adequacy and suitability of the materials used in the reactor pressure vessel and its appurtenances with respect to mechanical and physical properties, effects of irradiation, corrosion resistance, and fabricability; special processes used in the manufacture and fabrication of components; special nondestructive examination methods; special controls and processes for welding ferritic and austenitic stainless steels; fracture toughness tests for the ferritic materials; the reactor vessel surveillance program; and the tensile and fracture toughness tests performed on the reactor vessel studs and other fasteners. An evaluation of the impact of the TPBARs on the reactor vessel materials and surveillance program due to changes in reactor coolant chemistry and reactor vessel fluence was performed and is summarized in Section 2.5.3 of this document.

SRP 5.3.2
Pressure-Temperature Limits: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a; 10 CFR 50, Appendix A, GDC 1, 14, 31 and 32; 10 CFR 50, Appendix G; and 10 CFR 50, Appendix H. They specifically deal with the pressure-temperature limits imposed on the reactor coolant pressure boundary during operation and tests to assure adequate safety margins of structural integrity for the ferritic components of the boundary. Branch Technical Position – MTEB 5-2 provides a summary of the requirements. An evaluation of the impact of the TPBARs on the pressure-temperature limits due to changes in reactor vessel fluence was performed and is summarized in Section 2.5.3 of this document.

SRP 5.3.3
Reactor Vessel Integrity: This section is a special summary review of all factors relating to the integrity of the reactor vessel. These areas are reviewed separately in accordance with SRP Sections 5.2.3, 5.2.4, 5.3.1, and 5.3.2. The acceptance criteria and potential TPBAR impact are discussed above in regard to those sections.

SRP 5.4.1.1
Pump Flywheel Integrity (PWR): The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a and 10 CFR 50, Appendix A, GDC 1 and 4. They specifically deal with the specification of suitable material for the flywheel, fracture toughness testing, preservice inspection, design of the flywheel (to withstand normal conditions, anticipated transients, design basis loss of coolant accident, and safe shutdown earthquake without loss of structural integrity), overspeed testing, and the inservice inspection program. The incorporation of TPBARs will not impact the design or operation of the flywheel.

SRP 5.4.2.1
Steam Generator Materials: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 1, 14, 15 and 31, and 10 CFR 50, Appendix B. They specifically deal with the selection and fabrication of materials for the steam generator, the steam generator design as it pertains to avoiding crevice areas, compatibility of the tubing with the primary and secondary coolants, and the adequacy of the design to allow cleanup of deposits on the secondary side. The design and materials of the steam generator will not be changed for the incorporation of TPBARs. The effects of the TPC on reactor coolant chemistry were evaluated (see Section 3.8.3 for details). In the worst case for normal operation, where two TPBARs are breached, the reactor coolant chemistry remains well within the design limits for impurities and particulates. Given that the reactor coolant chemistry is not significantly affected, and the materials have not changed, there is no impact due to the TPC on this SRP.

SRP 5.4.2.2
Steam Generator Tube Inservice Inspection: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 32, and specifically deal with the accessibility of the steam generator tubes for inservice inspection and the proposed inspection program. The incorporation of the TPBARs does not cause a change in the steam generator design, therefore the tubes remain accessible and the inspection program is not impacted.

SRP 5.4.7
Residual Heat Removal (RHR) System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4, 5, 19 and 34. Those criteria which deal with the seismic design of systems, structures and components which could adversely impact the RHR system, the dynamic effects associated with flow instabilities and loads, the sharing of structures between units, isolation requirements, pressure relief requirements, pump protection requirements, test requirements, and operational procedures are not impacted by the presence of the TPBARs because the design of the RHR system is not changing. The acceptance criteria related to the adequacy of the water supply

for the auxiliary feedwater system requires that sufficient inventory is available to permit operation at hot shutdown for at least 4 hours, followed by cooldown to the conditions permitting operation of the RHR system. The acceptance criteria related to the functional capability of the RHR system requires the system to be capable of bringing the reactor to cold shutdown, with only offsite or onsite power available, within a reasonable period of time following shutdown, assuming the most limiting single failure. A qualitative evaluation of the impact of the TPBARs on the RHR system is summarized in Section 2.5.4 of this document.

SRP 5.4.11
Pressurizer Relief Tank: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2 and 4. They specify that a failure or malfunction of the system will not have an adverse effect on equipment necessary to bring the plant to a safe shutdown condition, to prevent accidents, or to mitigate the consequences of an accident. Specifically, the system shall not endanger safety-related systems in the event of a seismic event, and failure of the system shall not result in missiles or adverse environmental conditions that could damage safety-related systems or components. The system design has not been changed as a result of incorporation of TPBARs, therefore, the criteria are still met.

SRP 5.4.12
Reactor Coolant System High Point Vents: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a; 10 CFR 50.46; and 10 CFR 50, Appendix A, GDC 1, 14 and 30. They specifically deal with the capability of the vent system to remove noncondensible gases from the primary coolant system with a minimal probability of inadvertent or spurious actuation. The design of the vent system is not being changed for the incorporation of the TPBARs, therefore the criteria are still met.

2.5.2 COLD OVERPRESSURE MITIGATION (SRP 5.2.2) EVALUATION

2.5.2.1 Introduction

Nuclear power plants licensed in the United States are required to have a means of providing for the prevention of the violation of the ASME Section XI Appendix G reactor vessel pressure limit. This is of concern at cold conditions (generally below 350°F reactor coolant temperature), when the plant is undergoing heatup or cooldown operation. On the reference plant, the means of ensuring that the reactor coolant system (RCS) pressure is maintained below the Appendix G limit is via the use of either of the following means of pressure mitigation:

a. Pressurizer Power Operated Relief Valves (PORVs)

b. Residual Heat Removal (RHR) relief valves

The potential causes of violations of the Appendix G limit can be classified as resulting from one of the following causes:

a. Mass injection into the RCS

b. Heat injection into the RCS

The pressure mitigating system to prevent violations of the Appendix G limit is termed within Westinghouse as the "Cold Overpressure Mitigation System" (COMS). This system is known in the industry as the "Low Temperature Overpressure Protection System" (LTOPS) or "Cold Overpressure Protection System" (COPS).

The incorporation of a full complement of TPBARs in the plant would have an effect on the COMS operation. The remainder of this section will analyze this effect.

2.5.2.2 Methodology

The methodology used was to review what effect the TPBARs would have on the following:

a. Reactor Vessel Appendix G pressure limit

b. Mass input design basis

c. Heat input design basis

d. Pressurizer PORVs setpoint or capacity

e. RHR relief valves setpoint or capacity

f. Consequential pressure transients

If any of the above are changed as a result of the TPC, the consequences of this change would be quantified.

As part of this effort, a reanalysis of the COMS setpoints was not performed. At the present time, any COMS analysis has to be performed in compliance with an NRC-approved procedure such as that described in Reference 1. The analysis of record for the representative plant predates the approval of Reference 1, and therefore does not comply with it. Any cold overpressure reanalysis that would have to be performed would have to comply with methodology such as that described in Reference 1; this is true for any reason that requires a COMS reanalysis or re-evaluation, not just for the TPC. Therefore, this evaluation was done solely to determine the effects of the TPC on the COMS operation.

2.5.2.3 Significant Input Parameters and Assumptions

The incorporation of a full complement of TPBARs will affect the Appendix G pressure limit, due to the changes in the neutron fluence on the reactor vessel. These fluence changes are predominantly due to the radial power distribution associated with the TPC. Section 2.5.3 includes the Appendix G limits for the plant with the TPC and for the same reactor vessel for the reference plant without the TPC.

The reference plant and the TPC have Model F steam generators and the same RCS flows. Also, there are no changes to the RCS besides the fuel that would affect the thermal/hydraulic plant behavior.

The following assumptions were used in this evaluation:

It was assumed that the design basis operational behavior of the plant would not be revised as a result of incorporation of the TPBARs. This is based upon the assumptions of:

a. The TPBAR, inserted into the thimble, is designed to meet the requirements of a large four-loop Westinghouse reactor under Conditions I, II, III, and IV, defined in the Safety Analysis Report of the reference plant.

b. The incorporation of a full complement of TPBARs will not cause or create any new accident scenarios other than those currently identified in Regulatory Guide 1.70 and the NRC Standard Review Plan or add to the probability of existing scenarios.

c. There will be no plant power uprating (or derating) or RCS thermal-hydraulic parameter changes made to enhance reactor performance or operational flexibility for the TPC.

Therefore, it was assumed that the TPC design would not result in a change in the cold overpressure design basis transients or the resulting performance of the COMS. The PORV and RHR valve performance and flow capacities would not be affected, and the RCS pressure response during the Cold Overpressure design basis transients would not be affected. The only effect of TPBARs on Cold Overpressure would be the revision to the Appendix G limit curves and therefore a consequential revision of the pressurizer PORV setpoint programs.

2.5.2.4 Results

The effect of the TPBARs is to lower the Appendix G limit (Section 2.5.4). The following are the differences between the pressure limit for the TPC and the reference plant.

Temperature, °F	Pressure Limit, psig		Difference, psig
	TPC	Reference Plant	
60	471	472	-1
70	485	486	-1
80	501	502	-1
90	519	521	-2
100	540	542	-2
110	547	547	0
140	662	666	-4
150	706	710	-4
160	756	761	-5
170	800	800	0
180	800	800	0
190	800	800	0
200	800	800	0

The pressure limit was developed by the following method:

1. From the Appendix G limits defined in Section 2.5.3, subtract 74 psi. This is the pressure difference between the wide-range pressure transmitter and the reactor vessel beltline. Since the pressure transmitter will read a higher pressure than that at the reactor vessel beltline, and the transmitter reading is used to actuate the pressurizer PORV, the relevant pressure value is the one at the transmitter.

2. The maximum pressure limit to be considered is the lower of the adjusted Appendix G limit or the 800 psig maximum pressure defined to provide protection for the PORV discharge piping.

Based on the difference in the pressure limit between the TPC and the reference plant, the PORV setpoints would be affected by only about 5 psi barring any modifications necessitated by the use of the latest NRC-approved methodology.

2.5.2.5 Conclusions

As described above, the effect of the incorporation of a full complement of TPBARs on the COMS is minor, with the Appendix G limit being reduced by 5 psi or less over the pressure range of consideration. This would necessitate only about a 5 psi reduction in the pressurizer PORV setpoint program for the TPC, assuming no change in analysis methodology.

Since the Appendix G limits are affected and a PORV setpoint program revision is necessary, the following plant-specific documents would require revisions:

1. COMS setpoint analysis report.

2. Precautions, Limitations, and Setpoints document to include revised PORV setpoints.

3. Technical Specifications (or Pressure-Temperature Limits Report, PTLR) to show revised Appendix G limits and maximum allowable PORV setpoints.

2.5.2.6 Potential Plant Specific Analysis

COMS setpoints would need to be revised for any plant for which the Appendix G limits change. The latest NRC-approved methodology would have to be used.

2.5.2.7 References

1. WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", Rev. 2, January, 1996

2.5.3 REACTOR VESSEL INTEGRITY (SRP 5.3.1, 5.3.2, AND 5.3.3) EVALUATION

2.5.3.1 Introduction

Reactor vessel integrity is impacted by any changes in plant parameters that affect neutron fluence levels or temperature/pressure transients. The changes in neutron fluence resulting from the core design changes associated with the addition of the Tritium Production Burnable Absorber Rods (TPBARs) in the reference plant reactor vessel have been evaluated to determine the impact on reactor vessel integrity. This assessment included a review of the current material surveillance capsule withdrawal schedules, a comparison of the plant heatup and cooldown pressure-temperature limit curves generated with and without the effects of the TPBARs at the end of life (EOL), applicability of the Emergency Response Guideline (ERG) (Reference 2) limits, and a revision to the RT_{PTS} values used in the submittal to the NRC for meeting the requirements of 10 CFR 50.61 (Reference 3), known as the Pressurized Thermal Shock (PTS) Rule.

The most critical area, in terms of reactor vessel integrity, is the beltline region of the reactor vessel. The beltline region is defined in ASTM E185-82 (Reference 4) as "the irradiated region of the reactor vessel (shell material including weld regions and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions that are predicted to experience sufficient neutron damage to warrant consideration in the selection of surveillance material". Figure 2.5.3-1 identifies and indicates the location of all beltline region material of the reference plant reactor vessel.

2.5.3.2 Methodology

The reactor vessel integrity evaluation for the TPC includes the following objectives:

1. Determine the predicted EOL upper shelf energy (USE) for the reference plant and for the TPC. These calculations are consistent with Regulatory Guide 1.99 Revision 2.

2. Determine if a revision to the reactor vessel surveillance capsule removal schedule is necessary for the reference plant to reflect the changes in vessel fluences due to the TPC. These calculations are consistent with the recommended practices of ASTM E185-82 and meet the requirements of Appendix H of 10 CFR Part 50 (Reference 5).

3. Calculate adjusted reference temperature (ART) values, following the methods of Regulatory Guide 1.99, Revision 2, for all beltline material based upon projected fluence values at 32 effective full power years (EFPY) which is EOL. The calculation will be performed for two conditions: 1.) normal operation without TPBARs at 32 EFPY and 2.) normal operation with TPBARs at 32 EFPY. A comparison of the ART values as well as the pressure-temperature curves will be made to determine the effects of the TPBARs. Also, the Emergency Response Guidelines (ERG) pressure-temperature limits will be determined.

4. Calculate RT_{PTS} values for all beltline material in the reference plant reactor vessel based upon fluence values projected for the TPC at EOL (32 EFPY). The current PTS Rule, 10 CFR Part 50.61, will be used to ensure that the screening criteria is met.

Predicted EOL Upper Shelf Energy Calculations

The EOL USE can be predicted using the EOL 1/4T (1/4 Thickness) fluence projections and the copper content of the beltline material per Figure 2 of Regulatory Guide 1.99, Revision 2.

Per Regulatory Guide 1.99, Revision 2:

EOL 1/4T fluence = surface fluence * $e^{-0.24x}$, where x is the depth into the vessel wall (inches).

Surveillance Capsule Withdrawal Schedules

A surveillance capsule withdrawal schedule is developed to periodically remove surveillance capsules from the reactor vessel in order to effectively monitor the condition of the reactor vessel materials under actual operating conditions. ASTM E185-82 defines the recommended number of surveillance capsules and the recommended withdrawal schedule, based on the vessel material predicted transition temperature shifts (ΔRT_{NDT}). The surveillance capsule withdrawal schedule is in terms of effective full-power years (EFPY) of plant operation with a design life of 32 EFPY. Other factors that must be considered in establishing the surveillance capsule withdrawal schedule are the maximum fluence values at the vessel surface and 1/4-thickness (1/4T) location.

The first surveillance capsule is usually scheduled to be withdrawn early in the vessel life to verify the initial predictions of the surveillance material response to the actual radiation environment. It is generally removed when the predicted shift exceeds the expected scatter by sufficient margin to be measurable. Normally, the capsule with the highest lead factor (ratio of capsule fluence rate to peak fluence rate at pressure vessel surface) is withdrawn first. Early withdrawal also permits verification of the adequacy and conservatism of the reactor vessel pressure-temperature operational limits.

The withdrawal schedule of the remaining capsules is adjusted by the lead factor to obtain data at specific fluence values (i.e., peak EOL 1/4T and peak surface fluence values). Per ASTM E185-82, the four steps used for the development of a surveillance capsule withdrawal schedule are as follows:

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

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

3. Calculate the EFPY for the capsule to reach the peak vessel EOL fluence at the inside surface and 1/4T locations.

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

A surveillance capsule withdrawal schedule has been developed for the reference plant reactor vessel.

Heatup and Cooldown Pressure-Temperature Limits Curves

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements" (Reference 6) specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Boiler and Pressure Vessel Code forms the basis for these requirements. Appendix G to Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components" (Reference 7), contains the conservative methods of analysis.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ia}, for the metal temperature at that time. K_{Ia} is obtained from the reference fracture toughness curve, defined in Appendix G of the ASME Code, Section XI. The K_{Ia} curve is given by the following equation:

$$K_{Ia} = 26.78 + 1.233 * e^{(0.0145\,(T-RT_{NDT}+160))} \qquad \text{Equation 2.5.3-1}$$

where,

K_{Ia} = reference stress intensity factor as a function of the metal temperature, T, and the metal reference nil-ductility temperature, RT_{NDT}

Therefore, the governing equation for the heatup-cooldown analysis is defined in ASME Section XI, Appendix G as follows:

$$C * K_{Im} + K_{It} < K_{Ia} \qquad \text{Equation 2.5.3-2}$$

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress,

K_{It} = stress intensity factor caused by the thermal gradients,

K_{Ia} = function of temperature relative to the RT_{NDT} of the material, and

C = 2.0 for Level A and Level B service limits or

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical.

At any time during the heatup or cooldown transient, the allowable value K_{Ia} is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses

resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From Equation 2.5.3-2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT (temperature) developed during cooldown results in a higher allowable value of K_{Ia} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in allowable value K_{Ia} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the allowable value K_{Ia} for the 1/4T crack during heatup is lower than the allowable value K_{Ia} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower allowable K_{Ia} values do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside

surface is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

10 CFR Part 50, Appendix G addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure, which is 621 psig for the reference plant.

Calculation of Adjusted Reference Temperature:

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression.

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad \text{Equation 2.5.3-3}$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code (Reference 8). If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = (CF) * f^{(0.28-0.10 \log f)} \qquad \text{Equation 2.5.3-4}$$

where,

CF = Chemistry factor from tables in Reference 1, and

f = Neutron fluence (10^{19} n/cm^2, E>1.0 MeV)

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(depth x)} = f_{surface} * e^{(-0.24x)} \qquad \text{Equation 2.5.3-5}$$

Where x inches (vessel beltline thickness is 8.625 inches, Reference 9) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 2.5.3-4 to calculate the ΔRT_{NDT} at the specific depth. The resultant ΔRT_{NDT} is then placed in Equation 2.5.3-3 to calculate the adjusted reference temperature (ART).

The resulting ART values were calculated using the beltline material properties and the fluence projections for the reference plant (without the TPBARs) per Regulatory Guide 1.99, Revision 2. These ART values were then compared to those ART values calculated using the fluence projections for the TPC. Both sets of ART values were used in the generation of the heatup and cooldown curves at EOL (32 EFPY). These curves were subsequently reviewed to determine the effects of the TPC on the pressure-temperature curves.

Pressurized Thermal Shock (PTS)

A limiting condition on reactor vessel integrity known as Pressurized Thermal Shock (PTS) may 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, and
- 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 pressurized thermal shock (Reference 10). It established screening criterion on pressurized water reactor (PWR) vessel embrittlement as measured by the nil-ductility reference temperature, termed RT_{PTS}. RT_{PTS} screening criteria values were set (using conservative fracture mechanics analysis techniques) for beltline axial welds and plates or forgings and for beltline circumferential weld seams for end-of-life plant operation. All PWR vessels in the United States have been required to evaluate vessel embrittlement in accordance with the criteria through end-of-life.

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 (Reference 3). This amendment makes the procedure for calculating RT_{PTS} values consistent with the methods given in Regulatory Guide 1.99, Revision 2.

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 shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material. ...This assessment must be updated whenever there is significant change in projected values of RT_{PTS}, or upon a request for a change in the expiration date for operation of the facility" (Reference 3). Note that changes to RT_{PTS} values are considered significant if either the previous value or the current value, or both values, exceed the screening criterion prior to expiration of the operating license, including any renewed term. The RT_{PTS} values must be calculated based on the methodology specified in this rule. The submittal must include:

    1. the bases for the projection (including any assumptions regarding core loading patterns), and

    2. copper and nickel content and fluence values used in the calculations for each beltline material. (If these values differ from those previously submitted to NRC, justification must be provided.)

- The RT_{PTS} (measure of fracture resistance) screening criterion values for the beltline region are:

    270°F for plates, forgings and axial welds, and

    300°F for circumferential weld materials.

- All values of RT_{PTS} must be verified to be bounding values for the specific reactor vessel. In doing this each plant should consider plant-specific information that could affect the level of embrittlement.

- Plant-specific PTS safety analyses are required before a plant is within three years of reaching the screening criterion, including analyses of alternatives to minimize the PTS concern.

- NRC approval for operation beyond the screening criterion is required.

In the PTS Rule, the NRC Staff has selected a conservative and uniform method for determining plant-specific values of RT_{PTS} at a given time. For the purpose of comparison with the screening criteria, the value of RT_{PTS} for the reactor vessel must be calculated for each weld and plate or forging in the beltline region by:

$$RT_{PTS} = RT_{NDT(u)} + M + \Delta RT_{PTS} \qquad \text{Equation 2.5.3-6}$$

where,

ΔRT_{PTS} = $\Delta RT_{NDT} = (CF) * f^{(0.28 - 0.10 * \log f)}$

$RT_{NDT(u)}$ = Initial reference temperature (RT_{NDT}) of the unirradiated material

M = Margin to be added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence, and calculational procedures.

$$M = 2\sqrt{\sigma_\Delta^2 + \sigma_i^2}\ ^\circ F$$

σ_i = 0°F when I (Initial RT_{NDT}) is a measured value

σ_i = 17°F when I (Initial RT_{NDT}) is a generic value

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

σ_Δ not to exceed $0.5 * \Delta RT_{PTS}$

f = Neutron fluence (10^{19} n/cm^2, E > 1.0 MeV) at the clad/base metal interface

CF = Chemistry factor from tables (Reference 1) for welds and for base metal (plates and forgings). If plant-specific surveillance data has been deemed credible per Regulatory Guide 1.99, Revision 2, it may be considered in the calculation of the chemistry factor.

RT_{PTS} values have been calculated for all of the beltline materials in the reference plant reactor vessel for both conditions (with and without the incorporation of the TPBARs).

Emergency Response Guideline (ERG) Limits

Emergency Response Guideline (ERG) pressure-temperature limits were developed in order to establish guidance for operator action in the event of an emergency situation, 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} which exceeds 250°F for a longitudinal flaw or 300°F for a circumferential flaw, plant-specific ERG pressure-temperature limits must be developed.

The RT_{NDT} values at the end-of-life (32 EFPY) were calculated using both the current reference plant fluence projections and the increased fluence projections for the TPC to determine if the applicable ERG category has changed.

2.5.3.3 Input Parameters and Assumptions

Material Data

Before performing the assessment of reactor vessel integrity for the core design changes associated with the incorporation of a full complement of TPBARs for the reference plant, a review of the latest plant-specific material properties was performed. Material property values were derived from vessel fabrication test certificate results and any subsequent chemical analyses that may have been performed on the surveillance materials from the reference plant surveillance program (Reference 11). Fast neutron irradiation-induced changes in the tensile, fracture, and impact properties of reactor vessel materials are largely dependent on chemical composition, particularly in the copper concentration.

The reference plant reactor vessel pertinent chemical and mechanical properties were obtained from the most recent Capsule and PTS WCAP Reports (References 12, 13, and 14). A summary of the pertinent chemical and mechanical properties of the beltline region forgings and weld metal used in this evaluation is given in Table 2.5.3-1.

Fluence Projections

The TPC equilibrium core defined in Section 2.4.3, results in a calculated maximum neutron flux (E > 1.0 MeV) of 2.06×10^{10} n/cm^2-s at the pressure vessel inner radius. This value may be compared with the design basis neutron flux of 3.41×10^{10} n/cm^2-s and the cycle specific flux levels of 2.42×10^{10}, 1.93×10^{10}, 1.91×10^{10}, and 1.74×10^{10} n/cm^2-s in the reference plant for fuel cycles 1 through 4, respectively. The increased flux at the reactor pressure vessel for the TPC, relative to cycles 2 through 4 for the reference plant, is predominantly due to the placement of low burnup fuel on the periphery of the TPC.

Evaluations of the projected end of life (32 EFPY) peak vessel fluence based on the plant specific calculations were provided in conjunction with the analysis of reactor vessel surveillance capsule Y performed in 1995 and documented in Reference 12. In that WCAP, the projected end of life fluence (E > 1.0 MeV) was given as 2.03×10^{19} n/cm^2. This value was based on the assumption that the average neutron flux from cycles 1 through 4 (2.01×10^{10} n/cm^2-s) would be applicable over the remaining plant lifetime. If the TPC core were inserted at the end of Cycle 4 and remained in place for the remainder of plant lifetime yielding a peak flux of

2.06×10^{10} n/cm^2-s, the projected end of life fluence would increase to 2.06×10^{19} n/cm^2. This represents an increase of less than 1.5%; a value well within the uncertainty associated with the projections.

Since vessel fluence evaluations are required by the NRC to be performed on a best estimate rather than bounding basis, a re-evaluation of plant specific fluence values and the resultant impact on reactor vessel integrity would be required at the time of implementation of a TPC at any plant.

These fluence values were used to calculate the end-of-life transition temperature shift (EOL ΔRT_{NDT}) for development of the surveillance capsule withdrawal schedules, adjusted reference temperature (ART) values for determining the generation of new heatup and cooldown curves, RT_{PTS} values and ERG limits.

2.5.3.4 Acceptance Criteria

Surveillance Capsule Removal Schedule

The proposed surveillance capsule removal schedules developed for the reference plant following the implementation of the TPC shall meet the requirements of ASTM-185-82. A satisfactory number of surveillance capsules shall 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

New heatup and cooldown curves shall be generated for normal operation before and after implementation of the TPC at EOL (32 EFPY) to determine if significant reduction in allowable pressure occurs between the two sets of curves (i.e., curves for reactor vessel with and without TPBARs).

ERG Limits

The ERG limits shall be known in order to establish guidelines for operator action in the event of an emergency situation, such as a PTS event. The limits were obtained from the Emergency Response Guidelines - Revision 1B, and range from Category I (i.e., least severe) to Category III b (most severe).

Pressurized Thermal Shock (PTS)

The RT_{PTS} values shall be determined for the reference plant reactor vessel with and without the TPBARs for all beltline materials. These RT_{PTS} shall not exceed the screening criteria of the PTS Rule (Reference 3). Specifically, the RT_{PTS} values of the base metal (plates or forgings) shall not exceed 270°F, while the circumferential weld metal RT_{PTS} values shall not exceed 300°F through the EOL (32 EFPY). In addition, the RT_{PTS} values for both conditions were compared to determine the amount of increase due to the TPBARs.

2.5.3.5 Results

An evaluation of the impact of the TPC on the reactor vessel integrity was performed for the reference plant. Neutron fluence changes and other relevant system parameters were considered in the evaluation.

Surveillance Capsule Withdrawal Schedule

The resulting surveillance capsule withdrawal schedule is based on ASTM E185-82. Per ASTM E185-82, the withdrawal of a capsule is to be scheduled at the nearest vessel refueling outage to the calculated EFPY established for the particular surveillance capsule withdrawal. It was determined that the current reference plant surveillance capsule removal schedule would not require revision based on the 1.5 % increase in the projected vessel fluence due to the TPC. The current removal schedule is presented in Table 2.5.3-2.

Applicability of Heatup and Cooldown Pressure-Temperature Limit Curves

A review of the applicable heatup and cooldown curves showed a decrease in allowable pressure ranging from 1 to 30 psi due to the 1.5 % increase in the projected vessel fluence for the TPC. The delta pressure increased as a function of rising temperature, starting at 60°F. This change difference in temperature and the rate of change occurs for both heatup and cooldown conditions.

The chemical properties of the beltline material were used to calculate chemistry factor values per Regulatory Guide 1.99, Revision 2 (see Table 2.5.3-3). Additionally, chemistry factor values were determined using credible surveillance capsule data obtained from previously tested capsules (see Table 2.5.3-4) per Regulatory Guide 1.99, Revision 2. These calculations are presented in Table 2.5.3-5. Lastly, a summary of all chemistry factors used in these calculations are presented in Table 2.5.3-6.

The peak clad/base metal interface fluence (E > 1.0 MeV, n/cm^2) values at end of life (i.e., 32 EFPY) used in the reactor vessel integrity calculations are presented in Table 2.5.3-7. The fluence values are given for a reactor vessel with and without TPBARs. As stated above, the fluence would increase 1.5% on the reference plant reactor vessel due to incorporation of a full complement of TPBARs.

Following, in Tables 2.5.3-8 through 2.5.3-11, are the ART calculations using the fluence projections with and without the effects from the TPBARs. The ART values calculated for the vessel with TPBARs are approximately 1°F greater than the ART values calculated for the vessel without TPBARs. The ART values used to generate the curves are presented in Table 2.5.3-12 and the resulting data points from the heatup and cooldown curves are presented in Tables 2.5.3-13 through 2.5.3-16. The actual heatup and cooldown curves are presented in Figures 2.5.3-1 and 2.5.3-2.

ERG Limits

The EOL PTS calculations are the same as those used to determine the limiting EOL inside surface RT_{PTS} for the beltline materials. Therefore, for the reference plant, the peak inside surface RT_{PTS} value at EOL was calculated to be 126°F for the reactor vessel fluence without effects from the TPBARs and 127°F for the reactor vessel fluence with effects from the TPBARs (see Tables 2.5.3-18 and 2.5.3-19). These results are within the Category I criteria as provided in Table 2.5.3-17. Therefore, the reference plant will remain in ERG Pressure Temperature Limit Category I, which is the most favorable, after installation of TPBARs. Based on these results, there would be no need to change the current ERGs due to plant operation with the TPC in the reactor vessel.

Pressurized Thermal Shock (PTS)

Calculations were performed using the latest procedures specified by the NRC in the PTS Rule. RT_{PTS} values were generated for all beltline region material of the reference plant reactor vessel for EOL (32 EFPY). Tables 2.5.3-18 and 2.5.3-19 provide a summary of the resulting RT_{PTS} values for all beltline region material for both conditions; the reactor vessel with and without TPBARs at 32 EFPY. The fluence values presented in Table 2.5.3-7 were used in these RT_{PTS} calculations.

All RT_{PTS} values remain below the NRC screening criteria values using the projected fluence values through 32 EFPY for the reference plant. Specifically, the most limiting RT_{PTS} value at 32 EFPY is 127°F and 126°F (Lower Shell Plate R8-1) for the reactor vessel with and without TPBARs, respectively.

2.5.3.6 Conclusions

It is concluded that the TPBAR program applied to the reference plant will not have a significant impact on the reactor vessel integrity based on the following reasons:

- The surveillance capsule withdrawal schedule has been developed for normal operation (i.e., TPBARs not in the reactor vessel). The 1.5% increase in fluence due to a full complement of TPBARs would not change the current schedule which is given in Table 2.5.3-1.

- The calculation of the ART for the reference plant shows an increase of 1°F in the limiting beltline material after incorporation of a full complement of TPBARs in the reactor vessel. In addition, the resulting heatup and cooldown curves showed minor incremental decreases in allowable pressure ranging from 1 to 30 psi. Note that the lower decreases in pressure occurred at the lower operating temperatures.

- The reference plant will remain in Category I of the ERG pressure-temperature limits after the operation with the TPC in the reactor vessel. Hence, it is concluded that there would not be an Emergency Response Guideline Category change due to the TPC.

- All RT_{PTS} values of the reference plant reactor vessel beltline material remain below the PTS screening criteria using projected fluence values through 32 EFPY. The most limiting RT_{PTS}

value at 32 EFPY is 127°F for the circumferential weld. Therefore, the incorporation of a full complement of TPBARs in the reference plant reactor vessel will have no significant impact on the RT_{PTS} values through 32 EFPY.

Based on these conclusions the reference plant Pressure-Temperature Limits Report (PTLR) and FSAR would need to be updated to reflect the change to the PTS value and include the updated pressure-temperature curves for the applicable EFPY.

2.5.3.7 Potential Plant-Specific Analytical Issues

The conclusions of this evaluation for the reference plant indicated that the 1.5% increase in projected vessel fluence due to the TPC will have minimal impact on reactor vessel/plant operation. However, if a plant that is close to the PTS screening criteria, or a plant with a small operating window during heatup and cooldown, or a plant with high copper/nickel content in the beltline materials was selected for the TPC, then the acceptance criteria may not be met.

2.5.3.8 References

1. Regulatory Guide 1.99, Revision 2, May 1988, "Radiation Embrittlement of Reactor Vessel Materials".
2. Emergency Response Guidelines - Revision 1B, Westinghouse Owners Group, 2/28/92.
3. CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", December 19, 1995. (PTS Rule).
4. ASTM E185-82, Annual Book of ASTM Standards, Section 12, Volume 12.02, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E706 (IF)[1]".
5. 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements", 1/1/90 Edition.
6. 10 CFR 50, Appendix G, "Fracture Toughness Requirements", Federal Register, Volume 60, No. 243, dated December 19,1995.
7. ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure".
8. Section III, Division 1 of the ASME B&PV Code, Paragraph NB-2331, "Materials for Vessels".
9. Reference Plant Drawing, "General Arrangement-Elevation".
10. 10 CFR 50, "Analysis of Potential Pressurized Thermal Shock Events", July 23, 1985.
11. WCAP-11381, "(Reference Plant) Reactor Vessel Radiation Surveillance Program", Non-Proprietary, L.R. Singer, April 1986.
12. WCAP-14532, "Analysis of Capsule Y From the (Reference Plant) Reactor Vessel Radiation Surveillance Program", Non-Proprietary, P.A. Grendys, et.al., February 1996.

13. WCAP-14534, "Evaluation of Pressurized Thermal Shock For (Reference Plant)", Non-Proprietary, P.A. Grendys, February 1996.

14. WCAP-14533, "(Reference Plant) Heatup and Cooldown Limit Curves for Normal Operation", Non-Proprietary, P.A. Grendys, February 1996.

Table 2.5.3-1 Chemical and Mechanical Properties of the Beltline Region Materials				
Material Description	**Cu (%)**	**Ni (%)**	**Chemistry Factor[a]**	**Initial RT_{NDT} (°F)[b]**
Closure Head Flange	--	--	N/A	10[c][d]
Vessel Flange	--	--	N/A	-60[c][d]
Intermediate Shell Plate R4-1	0.07	0.63	44	10
Intermediate Shell Plate R4-2	0.06	0.61	37	10
Intermediate Shell Plate R4-3	0.05	0.60	31	30
Lower Shell Plate R8-1	0.07	0.63	44	40
Lower Shell Plate B8628-1	0.05	0.59	31	50
Lower Shell Plate B8825-1	0.06	0.62	37	40
Longitudinal Welds	0.04	0.15	38.3	-10
Circumferential Weld	0.04	0.15	38.3	-30

<u>NOTES:</u>

(a) Chemistry Factors are calculated from Cu and Ni values per Regulatory Guide 1.99, Revision 2, Pos. 1.1 (Reference 1).

(b) Initial RT_{NDT} values are measured values.

(c) Closure head and vessel flange Initial RT_{NDT} values are used for considering flange requirements (Reference 6) for the heatup/cooldown curves.

(d) Per Reference 14.

Table 2.5.3-2
Reference Plant Surveillance Capsule Withdrawal Schedule

Capsule	Capsule Location (Degree)	Lead Factor	Withdrawal EFPY[a]	Fluence (n/cm^2)
U[b]	58.5	4.41	1.18	4.42×10^{18}[e]
Y[b]	241	4.09	4.83	1.13×10^{19}[e]
X	238.5	4.41	8.6	2.41×10^{19}[c]
W	121.5	4.41	12.2	3.43×10^{19}[d]
Z	301.5	4.41	Standby[f]	- -
V	61	4.09	Standby[f]	- -

<u>NOTES:</u>

(a) Effective Full Power Years (EFPY) from plant startup.

(b) Plant-specific evaluation.

(c) Approximately equal to the 38 EFPY inner vessel wall fluence.

(d) Approximately equal to the projected peak vessel fluence at 54 EFPY, not less than once or greater than twice the maximum EOL inner vessel wall fluence. Per ASTM E185-82, this capsule may be held without testing following withdrawal.

(e) Updated in Capsule Y dosimetry analysis, section 6.0 of WCAP-14532 (Reference 12).

(f) It is recommended that the "Standby" capsules be removed and stored at the same time Capsule W is removed.

Table 2.5.3-3 Interpolation of Chemistry Factors per Regulatory Guide 1.99, Revision 2 for Reference Plant		
Material	**Ni, wt%**	**Chemistry Factor, °F**
Intermediate Shell Plate R4-1 Given Cu wt % = 0.07	0.63	44
Intermediate Shell Plate R4-2 Given Cu wt % = 0.06	0.61	37
Intermediate Shell Plate R4-3 Given Cu wt % = 0.05	0.60	31
Lower Shell Plate R8-1 Given Cu wt % = 0.07	0.63	44
Lower Shell Plate B8628-1 Given Cu wt % = 0.05	0.59	31
Lower Shell Plate B8825-1 Given Cu wt % = 0.06	0.62	37
Weld Metal	0.00	24
Given Cu wt % = 0.04	0.15	38.3
	0.20	43

Table 2.5.3-4 Summary of Surveillance Capsule Fluence Values for Reference Plant		
Capsule	**EFPY**	**Capsule Fluence, n/cm²**
U	1.18	4.42×10^{18}
Y	4.83	1.13×10^{19}

Table 2.5.3-5 Calculation of Chemistry Factors Using Credible Surveillance Capsule (S/C) Data						
Material	**Capsule**	**Capsule f**	**FF**	**ΔRT_{NDT}, °F**	**$FF^*\Delta RT_{NDT}$**	**FF^2**
Lower Shell Plate B8628-1	U	0.422	0.760	2.12	1.61	0.578
(Longitudinal)	Y	1.13	1.03	5.76	5.96	1.07
Lower Shell Plate B8628-1	U	0.422	0.760	0.00	0.00	0.578
(Transverse)	Y	1.13	1.03	1.93	2.00	1.07
	SUM:				9.56	3.30
	$CF = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (9.56) \div (3.30) = 2.9°F$					
Circumferential Weld 101-171	U	0.422	0.760	0.00	0.00	0.578
	Y	1.13	1.03	18.59	19.15	1.07
	SUM:				19.15	1.65
	$CF = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (19.15) \div (1.65) = 11.6°F$					

NOTES:

f = fluence $\div$ 10^{19} n/cm^2

FF = fluence factor = $f^{(0.28 - 0.1^*\log f)}$

$CF = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2)$

Table 2.5.3-6 Summary of Chemistry Factor (CF) Values for Beltline Material		
Beltline Materials	**Chemistry Factor, °F (Per Pos. 1.1 of RG 1.99 R.2)**	**Chemistry Factor, °F (Per Pos. 2.1 of RG 1.99 R.2)**
Inter. Shell Plate R4-1	44.0	- -
Inter. Shell Plate R4-2	37.0	- -
Inter. Shell Plate R4-3	31.0	- -
Lower Shell Plate R8-1	44.0	- -
Lower Shell Plate B8628-1	31.0	2.9
Lower Shell Plate B8825-1	37.0	- -
Inter. Shell Long. Weld 101-124A	38.3	- -
Inter. Shell Long. Weld 101-124B	38.3	- -
Inter. Shell Long. Weld 101-124C	38.3	- -
Lower Shell Long. Weld 101-142A	38.3	- -
Lower Shell Long. Weld 101-142B	38.3	- -
Lower Shell Long. Weld 101-142C	38.3	- -
Circumferential Weld 101-171	38.3	11.6

Table 2.5.3-7 Peak Fluence (E>1.0 MeV) Value on the Pressure Vessel Clad/Base Metal Interface	
Case	**Fluence (n/cm^2, E > 1.0 MeV)**
w/o TPBARs	2.03×10^{19}
w/ TPBARs	2.06×10^{19}

Table 2.5.3-8 Calculation of ART Values for the 1/4T Location @ 32 EFPY Using the Peak Fluence @ Clad/Base Metal Interface w/o TPBAR								
Material	CF (°F)	f @ 32 EFPY[4]	1/4T f[4]	1/4T FF	IRT$_{NDT}$[1] (°F)	Margin (°F)	ΔRT_{NDT}[3] (°F)	ART[2] (°F)
Inter. Shell Plate R4-1	44.0	2.03	1.210	1.053	10	34	46.3	90
Inter. Shell Plate R4-2	37.0	2.03	1.210	1.053	10	34	39.0	83
Inter. Shell Plate R4-3	31.0	2.03	1.210	1.053	30	32.6	32.6	95
Lower Shell Plate R8-1	44.0	2.03	1.210	1.053	40	34	46.3	120
Lower Shell Plate B8628-1	31.0	2.03	1.210	1.053	50	32.6	32.6	115
- Using S/C Data	2.9	2.03	1.210	1.053	50	3.1	3.1	56
Lower Shell Plate B8825-1	37.0	2.03	1.210	1.053	40	34	39.0	113
Inter. Shell Longitudinal Weld 101-124A	38.3	2.03	1.210	1.053	-10	40.3	40.3	71
Inter. Shell Longitudinal Weld 101-124B	38.3	2.03	1.210	1.053	-10	40.3	40.3	71
Inter. Shell Longitudinal Weld 101-124C	38.3	2.03	1.210	1.053	-10	40.3	40.3	71
Lower Shell Longitudinal Weld 101-142A	38.3	2.03	1.210	1.053	-10	40.3	40.3	71
Lower Shell Longitudinal Weld 101-142B	38.3	2.03	1.210	1.053	-10	40.3	40.3	71
Lower Shell Longitudinal Weld 101-142C	38.3	2.03	1.210	1.053	-10	40.3	40.3	71
Circumferential Weld 101-171	38.3	2.03	1.210	1.053	-30	40.3	40.3	51
- Using S/C Data	11.6	2.03	1.210	1.053	-30	12.2	12.2	-6

Notes:

1. Initial RT_{NDT} values are measured values.
2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)
3. $\Delta RT_{NDT} = CF * FF$
4. fluence = 10^{19} n/cm^2, E>1.0 MeV

Table 2.5.3-9
Calculation of ART Values for the 3/4T Location @ 32 EFPY
Using the Peak Fluence @ Clad/Base Metal Interface w/o TPBAR

Material	CF (°F)	f @ 32 EFPY[4]	3/4T f[4]	3/4T FF	IRT_{NDT}[1] (°F)	Margin (°F)	ΔRT_{NDT}[3] (°F)	ART[2] (°F)
Inter. Shell Plate R4-1	44.0	2.03	0.430	0.765	10	33.7	33.7	77
Inter. Shell Plate R4-2	37.0	2.03	0.430	0.765	10	28.3	28.3	67
Inter. Shell Plate R4-3	31.0	2.03	0.430	0.765	30	23.7	23.7	77
Lower Shell Plate R8-1	44.0	2.03	0.430	0.765	40	33.7	33.7	107
Lower Shell Plate B8628-1	31.0	2.03	0.430	0.765	50	23.7	23.7	97
- Using S/C Data	2.9	2.03	0.430	0.765	50	2.2	2.2	54
Lower Shell Plate B8825-1	37.0	2.03	0.430	0.765	40	28.3	28.3	97
Inter. Shell Longitudinal Weld 101-124A	38.3	2.03	0.430	0.765	-10	29.3	29.3	49
Inter. Shell Longitudinal Weld 101-124B	38.3	2.03	0.430	0.765	-10	29.3	29.3	49
Inter. Shell Longitudinal Weld 101-124C	38.3	2.03	0.430	0.765	-10	29.3	29.3	49
Lower Shell Longitudinal Weld 101-142A	38.3	2.03	0.430	0.765	-10	29.3	29.3	49
Lower Shell Longitudinal Weld 101-142B	38.3	2.03	0.430	0.765	-10	29.3	29.3	49
Lower Shell Longitudinal Weld 101-142C	38.3	2.03	0.430	0.765	-10	29.3	29.3	49
Circumferential Weld 101-171	38.3	2.03	0.430	0.765	-30	29.3	29.3	29
- Using S/C Data	11.6	2.03	0.430	0.765	-30	8.9	8.9	-12

Notes:

1. Initial RT_{NDT} values are measured values.
2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)
3. ΔRT_{NDT} = CF * FF
4. fluence = 10^{19} n/cm^2, E>1.0 MeV

Table 2.5.3-10 Calculation of ART Values for the 1/4T Location @ 32 EFPY Using the Peak Fluence @ Clad/Base Metal Interface w/TPBAR								
Material	CF (°F)	f @ 32 EFPY[4]	1/4T f[4]	1/4T FF	IRT_{NDT}[1] (°F)	Margin (°F)	ΔRT_{NDT}[3] (°F)	ART[2] (°F)
Inter. Shell Plate R4-1	44.0	2.06	1.228	1.057	10	34	46.5	91
Inter. Shell Plate R4-2	37.0	2.06	1.228	1.057	10	34	39.1	83
Inter. Shell Plate R4-3	31.0	2.06	1.228	1.057	30	32.8	32.8	96
Lower Shell Plate R8-1	44.0	2.06	1.228	1.057	40	34	46.5	121
Lower Shell Plate B8628-1	31.0	2.06	1.228	1.057	50	32.8	32.8	116
- Using S/C Data	2.9	2.06	1.228	1.057	50	3.1	3.1	56
Lower Shell Plate B8825-1	37.0	2.06	1.228	1.057	40	34	39.1	113
Inter. Shell Longitudinal Weld 101-124A	38.3	2.06	1.228	1.057	-10	40.5	40.5	71
Inter. Shell Longitudinal Weld 101-124B	38.3	2.06	1.228	1.057	-10	40.5	40.5	71
Inter. Shell Longitudinal Weld 101-124C	38.3	2.06	1.228	1.057	-10	40.5	40.5	71
Lower Shell Longitudinal Weld 101-142A	38.3	2.06	1.228	1.057	-10	40.5	40.5	71
Lower Shell Longitudinal Weld 101-142B	38.3	2.06	1.228	1.057	-10	40.5	40.5	71
Lower Shell Longitudinal Weld 101-142C	38.3	2.06	1.228	1.057	-10	40.5	40.5	71
Circumferential Weld 101-171	38.3	2.06	1.228	1.057	-30	40.5	40.5	51
- Using S/C Data	11.6	2.06	1.228	1.057	-30	12.3	12.3	-5

Notes:

1. Initial RT_{NDT} values are measured values.
2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)
3. $\Delta RT_{NDT} = CF * FF$
4. fluence = 10^{19} n/cm^2, E>1.0 MeV

Table 2.5.3-11 Calculation of ART Values for the 3/4T Location @ 32 EFPY Using the Peak Fluence @ Clad/Base Metal Interface w/ TPBAR								
Material	CF (°F)	f @ 32 EFPY[4]	3/4T f[4]	3/4T FF	IRT_{NDT}[1] (°F)	Margin (°F)	ΔRT_{NDT}[3] (°F)	ART[2] (°F)
Inter. Shell Plate R4-1	44.0	2.06	0.436	0.769	10	33.8	33.8	78
Inter. Shell Plate R4-2	37.0	2.06	0.436	0.769	10	28.5	28.5	67
Inter. Shell Plate R4-3	31.0	2.06	0.436	0.769	30	23.8	23.8	78
Lower Shell Plate R8-1	44.0	2.06	0.436	0.769	40	33.8	33.8	108
Lower Shell Plate B8628-1	31.0	2.06	0.436	0.769	50	23.8	23.8	98
- Using S/C Data	2.9	2.06	0.436	0.769	50	2.2	2.2	54
Lower Shell Plate B8825-1	37.0	2.06	0.436	0.769	40	28.5	28.5	97
Inter. Shell Longitudinal Weld 101-124A	38.3	2.06	0.436	0.769	-10	29.5	29.5	49
Inter. Shell Longitudinal Weld 101-124B	38.3	2.06	0.436	0.769	-10	29.5	29.5	49
Inter. Shell Longitudinal Weld 101-124C	38.3	2.06	0.436	0.769	-10	29.5	29.5	49
Lower Shell Longitudinal Weld 101-142A	38.3	2.06	0.436	0.769	-10	29.5	29.5	49
Lower Shell Longitudinal Weld 101-142B	38.3	2.06	0.436	0.769	-10	29.5	29.5	49
Lower Shell Longitudinal Weld 101-142C	38.3	2.06	0.436	0.769	-10	29.5	29.5	49
Circumferential Weld 101-171	38.3	2.06	0.436	0.769	-30	29.5	29.5	29
- Using S/C Data	11.6	2.06	0.436	0.769	-30	8.9	8.9	-12

Notes:

1. Initial RT_{NDT} values are measured values.
2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)
3. ΔRT_{NDT} = CF * FF
4. fluence = 10^{19} n/cm^2, E>1.0 MeV

Table 2.5.3-12 Comparison of Heatup & Cooldown Curve ART Values		
EFPY	¼ T Limiting ART, °F	¾ T Limiting ART, °F
32 (w/o TPBAR)	120	107
32 (w/TPBAR)	121	108

Table 2.5.3-13
32 EFPY Heatup Curve Data Points, w/o TPBAR
(without Uncertainties for Instrumentation Errors)

60 Heatup		Critical. Limit		100 Heatup		Critical. Limit		Leak Test Limit	
T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig
60	0	253	0	60	0	253	0	231	2000
60	528	253	553	60	494	253	494	253	2485
65	528	253	543	65	494	253	494		
85	528	253	535	85	494	253	494		
90	528	253	530	90	494	253	494		
95	528	253	528	95	494	253	494		
100	528	253	528	100	494	253	494		
105	528	253	531	105	494	253	494		
110	531	253	535	110	494	253	494		
115	535	253	541	115	494	253	494		
120	541	253	549	120	494	253	494		
125	549	253	558	125	494	253	496		
130	558	253	569	130	496	253	500		
135	569	253	582	135	500	253	505		
140	582	253	596	140	505	253	511		
145	596	253	611	145	511	253	519		
150	611	253	628	150	519	253	528		
155	628	253	647	155	528	253	539		
160	647	253	667	160	539	253	551		
165	667	253	689	165	551	253	565		
170	689	253	713	170	565	253	580		
175	713	253	738	175	580	253	597		
180	738	253	766	180	597	253	615		
185	766	253	796	185	615	253	635		
190	796	253	828	190	635	253	658		

Table 2.5.3-13 (Continued)
32 EFPY Heatup Curve Data Points, w/o TPBAR

(without Uncertainties for Instrumentation Errors)

60 Heatup		Critical. Limit		100 Heatup		Critical. Limit		Leak Test Limit	
T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig
195	828	253	863	195	658	253	682		
200	863	253	900	200	682	253	708		
205	900	253	940	205	708	253	736		
210	940	255	984	210	736	255	767		
215	984	260	1030	215	767	260	800		
220	1030	265	1080	220	800	265	836		
225	1080	270	1133	225	836	270	874		
230	1133	275	1191	230	874	275	916		
235	1191	280	1252	235	916	280	960		
240	1252	285	1318	240	960	285	1008		
245	1318	290	1389	245	1008	290	1060		
250	1389	295	1465	250	1060	295	1115		
255	1465	300	1546	255	1115	300	1175		
260	1546	305	1633	260	1175	305	1239		
265	1633	310	1726	265	1239	310	1307		
270	1726	315	1826	270	1307	315	1380		
275	1826	320	1932	275	1380	320	1459		
280	1932	325	2046	280	1459	325	1543		
285	2046	330	2167	285	1543	330	1633		
290	2167	335	2297	290	1633	335	1730		
295	2297	340	2435	295	1730	340	1833		
300	2435			300	1833	345	1943		
				305	1943	350	2061		
				310	2061	355	2186		
				315	2186	360	2320		
				320	2320	365	2462		
				325	2462				

Table 2.5.3-14
32 EFPY Cooldown Curve Data Points, w/o TPBAR
(without Uncertainties for Instrumentation Errors)

Steady State		20°F		40°F		60°F		100°F	
T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig
60	0	60	0	60	0	60	0	60	0
60	546	60	505	60	463	60	420	60	332
65	553	65	512	65	470	65	428	65	340
70	560	70	520	70	478	70	436	70	350
75	568	75	528	75	487	75	445	75	360
80	576	80	536	80	496	80	455	80	371
85	585	85	546	85	506	85	465	85	382
90	595	90	556	90	516	90	476	90	395
95	605	95	567	95	528	95	489	95	409
100	616	100	578	100	540	100	502	100	424
105	621	105	591	105	554	105	516	105	441
110	621	110	604	110	568	110	532	110	458
115	621	115	619	115	584	115	548	115	477
120	621	120	621	120	600	120	566	120	498
125	621	125	621	125	618	125	585	125	520
130	621	130	621	130	621	130	606	130	544
130	702	130	670	130	638	135	629	135	570
135	721	135	690	135	659	140	653	140	598
140	740	140	711	140	681	145	679	145	629
145	762	145	733	145	706	150	707	150	661
150	784	150	758	150	732	155	737	155	697
155	809	155	784	155	760	160	770	160	735
160	835	160	812	160	790	165	805	165	776
165	864	165	842	165	823	170	843	170	820
170	894	170	875	170	858	175	883	175	868
175	927	175	910	175	895	180	927	180	919
180	962	180	948	180	936	185	974	185	974

Table 2.5.3-14 (Continued)
32 EFPY Cooldown Curve Data Points, w/o TPBAR

(without Uncertainties for Instrumentation Errors)

Steady State		20°F		40°F		60°F		100°F	
T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig
185	999	185	988	185	980	190	1025	190	1034
190	1040	190	1032	190	1027	195	1079		
195	1083	195	1079	195	1077				
200	1130	200	1129						
205	1180								
210	1234								
215	1292								
220	1354								
225	1421								
230	1492								
235	1569								
240	1651								
245	1739								
250	1833								
255	1934								
260	2042								
265	2157								
270	2280								

Table 2.5.3-15 32 EFPY Heatup Curve Data Points, w/TPBAR (without Uncertainties for Instrumentation Errors)									
60 Heatup		Critical. Limit		100 Heatup		Critical. Limit		Leak Test Limit	
T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig
60	0	254	0	60	0	254	0	232	2000
60	526	254	552	60	491	254	491	254	2485
65	526	254	541	65	491	254	491		
85	526	254	533	85	491	254	491		
90	526	254	528	90	491	254	491		
95	526	254	526	95	491	254	491		
100	526	254	526	100	491	254	491		
105	526	254	529	105	491	254	491		
110	529	254	533	110	491	254	491		
115	533	254	539	115	491	254	491		
120	539	254	546	120	491	254	492		
125	546	254	556	125	492	254	494		
130	556	254	566	130	494	254	497		
135	566	254	579	135	497	254	502		
140	579	254	592	140	502	254	508		
145	592	254	607	145	508	254	516		
150	607	254	624	150	516	254	525		
155	624	254	642	155	525	254	535		
160	642	254	662	160	535	254	547		
165	662	254	684	165	547	254	561		
170	684	254	707	170	561	254	576		
175	707	254	733	175	576	254	592		
180	733	254	760	180	592	254	610		
185	760	254	789	185	610	254	630		
190	789	254	821	190	630	254	652		
195	821	254	856	195	652	254	676		
200	856	254	892	200	676	254	702		

Table 2.5.3-15 (Continued)
32 EFPY Heatup Curve Data Points, w/TPBAR
(without Uncertainties for Instrumentation Errors)

60 Heatup		Critical. Limit		100 Heatup		Critical. Limit		Leak Test Limit	
T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig
205	892	254	932	205	702	254	730		
210	932	255	975	210	730	255	760		
215	975	260	1020	215	760	260	792		
220	1020	265	1069	220	792	265	828		
225	1069	270	1122	225	828	270	866		
230	1122	275	1178	230	866	275	906		
235	1178	280	1239	235	906	280	950		
240	1239	285	1304	240	950	285	998		
245	1304	290	1374	245	998				
250	1374	295	1449	250	1048				
255	1449	300	1529	255	1103				
260	1529	305	1615	260	1162				
265	1615	310	1706	265	1224				
270	1706	315	1805	270	1292				
275	1805	320	1910	275	1364				
280	1910	325	2022	280	1442				
285	2022	330	2142	285	1525				
290	2142	335	2269	290	1614				
295	2269	340	2406	295	1709				
300	2406			300	1811				
				305	1919				
				310	2035				
				315	2159				
				320	2291				
				325	2432				

Table 2.5.3-16
32 EFPY Cooldown Curve Data Points, w/TPBAR
(without Uncertainties for Instrumentation Errors)

Steady State		20°F		40°F		60°F		100°F	
T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig
60	0	60	0	60	0	60	0	60	0
60	545	60	504	60	461	60	418	60	330
65	552	65	511	65	469	65	426	65	338
70	559	70	518	70	476	70	434	70	347
75	566	75	526	75	485	75	443	75	357
80	575	80	534	80	494	80	452	80	368
85	583	85	544	85	504	85	463	85	380
90	593	90	554	90	514	90	474	90	392
95	603	95	564	95	525	95	486	95	406
100	614	100	576	100	538	100	499	100	421
105	621	105	588	105	551	105	513	105	437
110	621	110	602	110	565	110	528	110	454
115	621	115	616	115	580	115	545	115	473
120	621	120	621	120	597	120	562	120	494
125	621	125	621	125	615	125	581	125	516
130	621	130	621	130	621	130	602	130	539
130	699	130	666	130	634	135	624	135	565
135	717	135	686	135	654	140	648	140	593
140	736	140	706	140	677	145	673	145	622
145	757	145	729	145	701	150	701	150	655
150	780	150	753	150	726	155	731	155	689
155	804	155	778	155	754	160	763	160	727
160	830	160	806	160	784	165	798	165	767
165	858	165	836	165	816	170	835	170	811
170	888	170	868	170	851	175	875	175	858
175	920	175	903	175	888	180	918	180	908
180	955	180	940	180	928	185	965	185	963

Table 2.5.3-16 (Continued)
32 EFPY Cooldown Curve Data Points, w/TPBAR
(without Uncertainties for Instrumentation Errors)

Steady State		20°F		40°F		60°F		100°F	
T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig	T, °F	P, psig
185	992	185	980	185	971	190	1015	190	1021
190	1031	190	1023	190	1017	195	1068		
195	1075	195	1069	195	1067				
200	1121	200	1119	200	1120				
205	1170								
210	1223								
215	1280								
220	1341								
225	1407								
230	1477								
235	1553								
240	1634								
245	1721								
250	1814								
255	1913								
260	2020								
265	2134								
270	2255								

Table 2.5.3-17 ERG Pressure-Temperature Limits*	
Applicable RT_{NDT} (ART) Value	**ERG Pressure-Temperature Limit Category**
$RT_{NDT} \leq 200°F$	Category I
$200°F < RT_{NDT} \leq 250°F$	Category II
$250°F < RT_{NDT} \leq 300°F$	Category IIIb

* Reference 2

Table 2.5.3-18 RT_{PTS} Calculations for Reference Plant Beltline Region Materials at 32 EFPY, w/o TPBAR							
Material	**f @ 32 EFPY[4]**	**FF**	**CF (°F)**	**ΔRT_{PTS}[3] (°F)**	**Margin (°F)**	**$RT_{NDT(U)}$[1] (°F)**	**RT_{PTS}[2] (°F)**
Inter. Shell Plate R4-1	2.03	1.19	44.0	52.4	34	10	96
Inter. Shell Plate R4-2	2.03	1.19	37.0	44.0	34	10	88
Inter. Shell Plate R4-3	2.03	1.19	31.0	36.9	34	30	101
Lower Shell Plate R8-1	2.03	1.19	44.0	52.4	34	40	126
Lower Shell Plate B8628-1	2.03	1.19	31.0	36.9	34	50	121
- Using S/C Data	2.03	1.19	2.9	3.5	3.5	50	57
Lower Shell Plate B8825-1	2.03	1.19	37.0	44.0	34	40	118
Inter. Shell Longitudinal Weld 101-124A	2.03	1.19	38.3	45.6	45.6	-10	81
Inter. Shell Longitudinal Weld 101-124B	2.03	1.19	38.3	45.6	45.6	-10	81
Inter. Shell Longitudinal Weld 101-124C	2.03	1.19	38.3	45.6	45.6	-10	81
Lower Shell Longitudinal Weld 101-142A	2.03	1.19	38.3	45.6	45.6	-10	81
Lower Shell Longitudinal Weld 101-142B	2.03	1.19	38.3	45.6	45.6	-10	81
Lower Shell Longitudinal Weld 101-142C	2.03	1.19	38.3	45.6	45.6	-10	81
Circumferential Weld 101-171	2.03	1.19	38.3	45.6	45.6	-30	61
- Using S/C Data	2.03	1.19	11.6	13.8	13.8	-30	-2

Notes:

1. Initial RT_{NDT} values are measured values.
2. $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)
3. $\Delta RT_{PTS} = CF * FF$
4. fluence = 10^{19} n/cm², E>1.0 MeV

Table 2.5.3-19 RT_{PTS} Calculations for Reference Plant Beltline Region Materials at 32 EFPY, w/TPBAR							
Material	f @ 32 EFPY[4]	FF	CF (°F)	ΔRT_{PTS}[3] (°F)	Margin (°F)	$RT_{NDT(U)}$[1] (°F)	RT_{PTS}[2] (°F)
Inter. Shell Plate R4-1	2.06	1.20	44.0	52.8	34	10	97
Inter. Shell Plate R4-2	2.06	1.20	37.0	44.4	34	10	88
Inter. Shell Plate R4-3	2.06	1.20	31.0	37.2	34	30	101
Lower Shell Plate R8-1	2.06	1.20	44.0	52.8	34	40	127
Lower Shell Plate B8628-1	2.06	1.20	31.0	37.2	34	50	121
- Using S/C Data	2.06	1.20	2.9	3.5	3.5	50	57
Lower Shell Plate B8825-1	2.06	1.20	37.0	44.4	34	40	118
Inter. Shell Longitudinal Weld 101-124A	2.06	1.20	38.3	46.0	46.0	-10	82
Inter. Shell Longitudinal Weld 101-124B	2.06	1.20	38.3	46.0	46.0	-10	82
Inter. Shell Longitudinal Weld 101-124C	2.06	1.20	38.3	46.0	46.0	-10	82
Lower Shell Longitudinal Weld 101-142A	2.06	1.20	38.3	46.0	46.0	-10	82
Lower Shell Longitudinal Weld 101-142B	2.06	1.20	38.3	46.0	46.0	-10	82
Lower Shell Longitudinal Weld 101-142C	2.06	1.20	38.3	46.0	46.0	-10	82
Circumferential Weld 101-171	2.06	1.20	38.3	46.0	46.0	-30	62
- Using S/C Data	2.06	1.20	11.6	13.9	13.9	-30	-2

Notes:

1. Initial RT_{NDT} values are measured values.
2. $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)
3. $\Delta RT_{PTS} = CF * FF$
4. fluence = 10^{19} n/cm², E>1.0 MeV

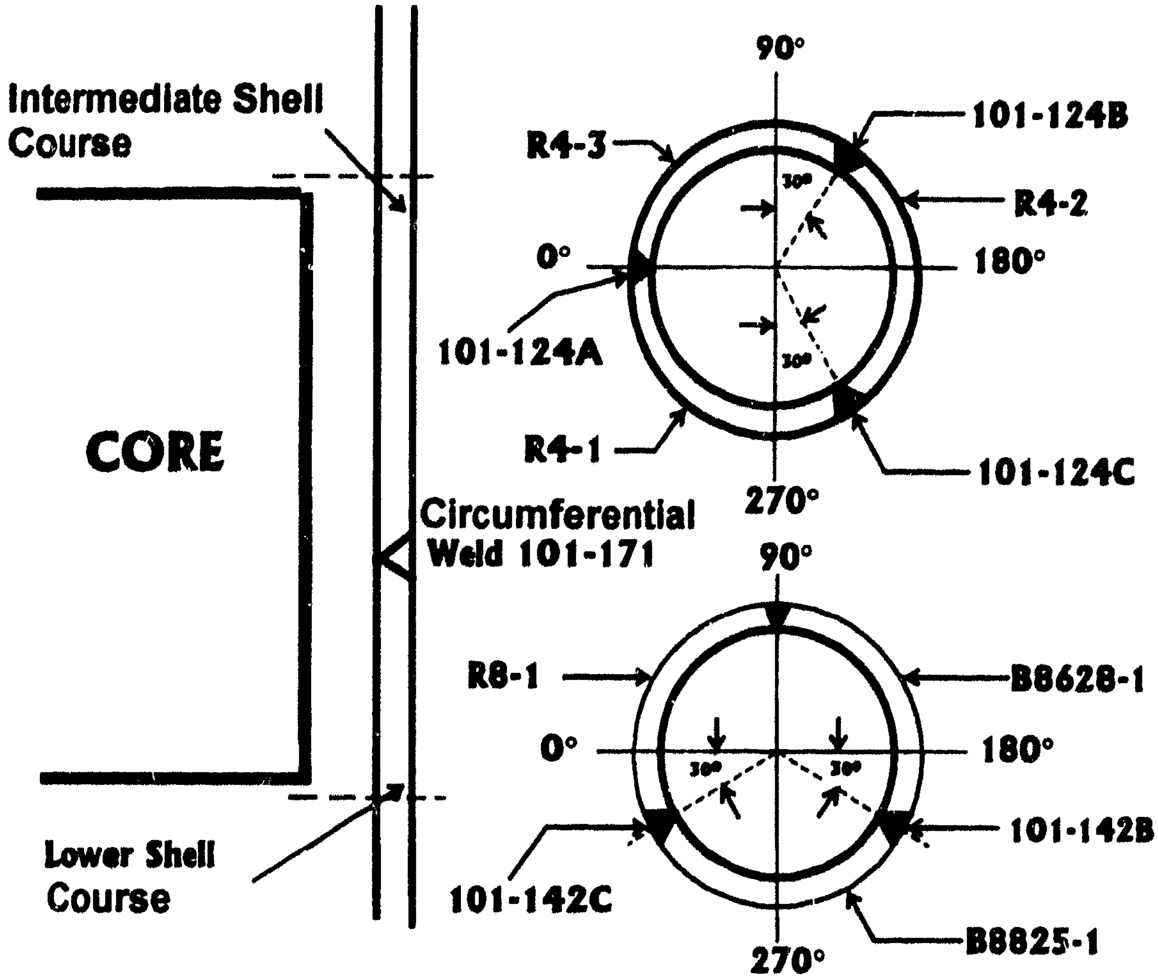

Figure 2.5.3-1
Identification and Location of Beltline Region Material for the Reference Plant Reactor Vessel (Reference 13)

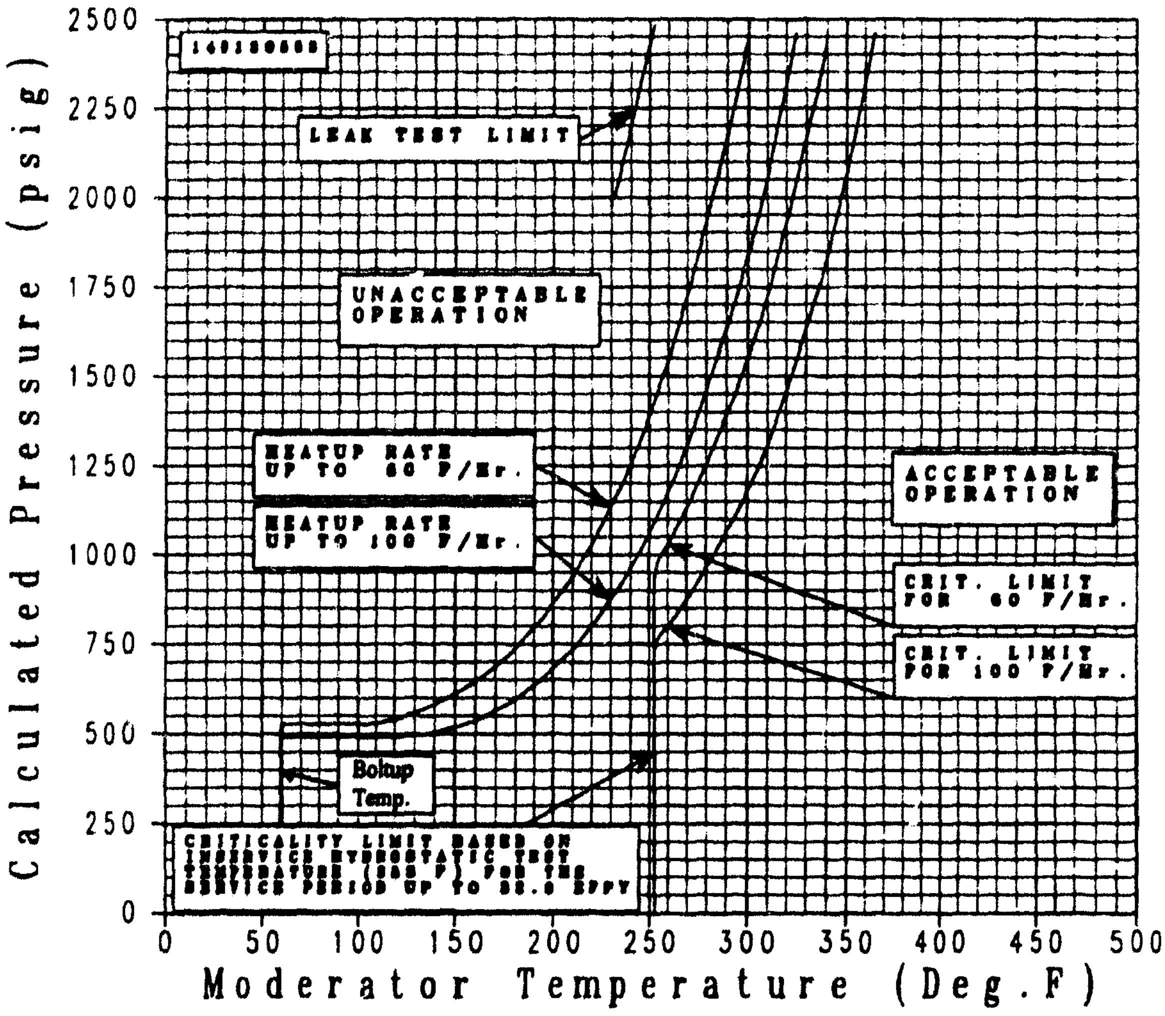

Figure 2.5.3-2
Heatup Curves Applicable to 32 EFPY w/o TPBAR
(without Uncertainties for Instrumentation Errors)

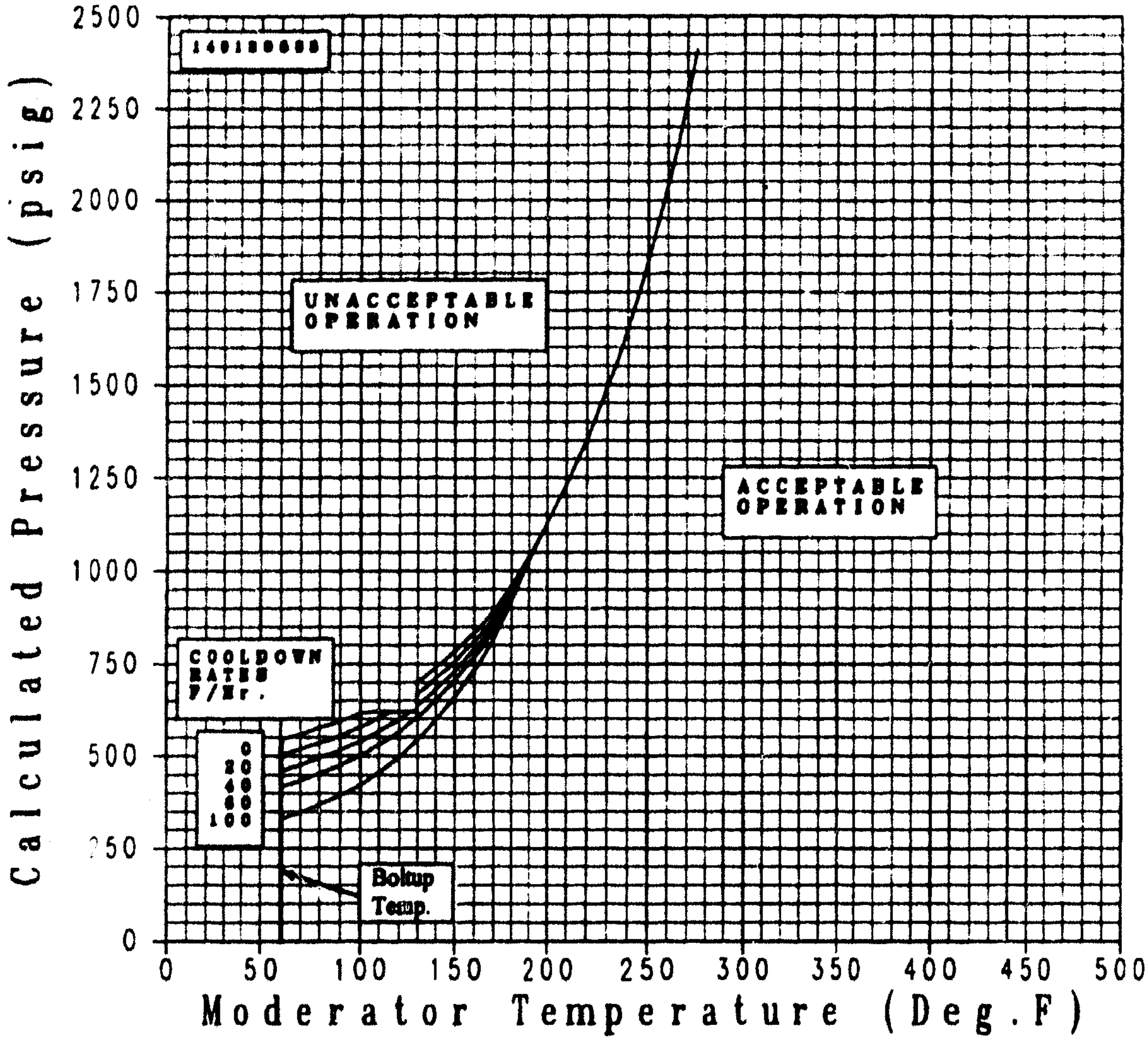

Figure 2.5.3-3
Cooldown Curves Applicable to 32 EFPY w/o TPBAR
(without Uncertainties for Instrumentation Errors)

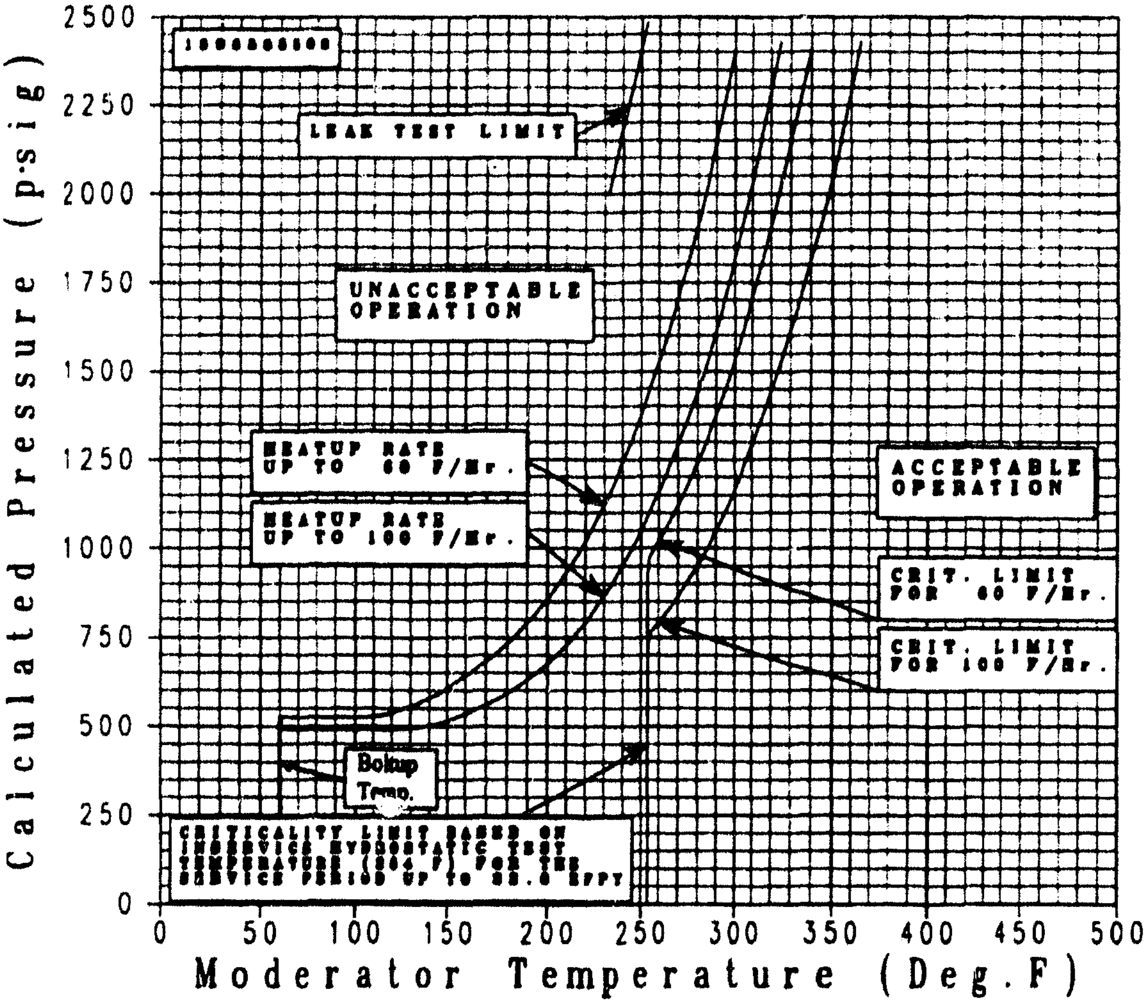

Figure 2.5.3-4
Heatup Curves Applicable to 32 EFPY w/TPBAR
(without Uncertainties for Instrumentation Errors)

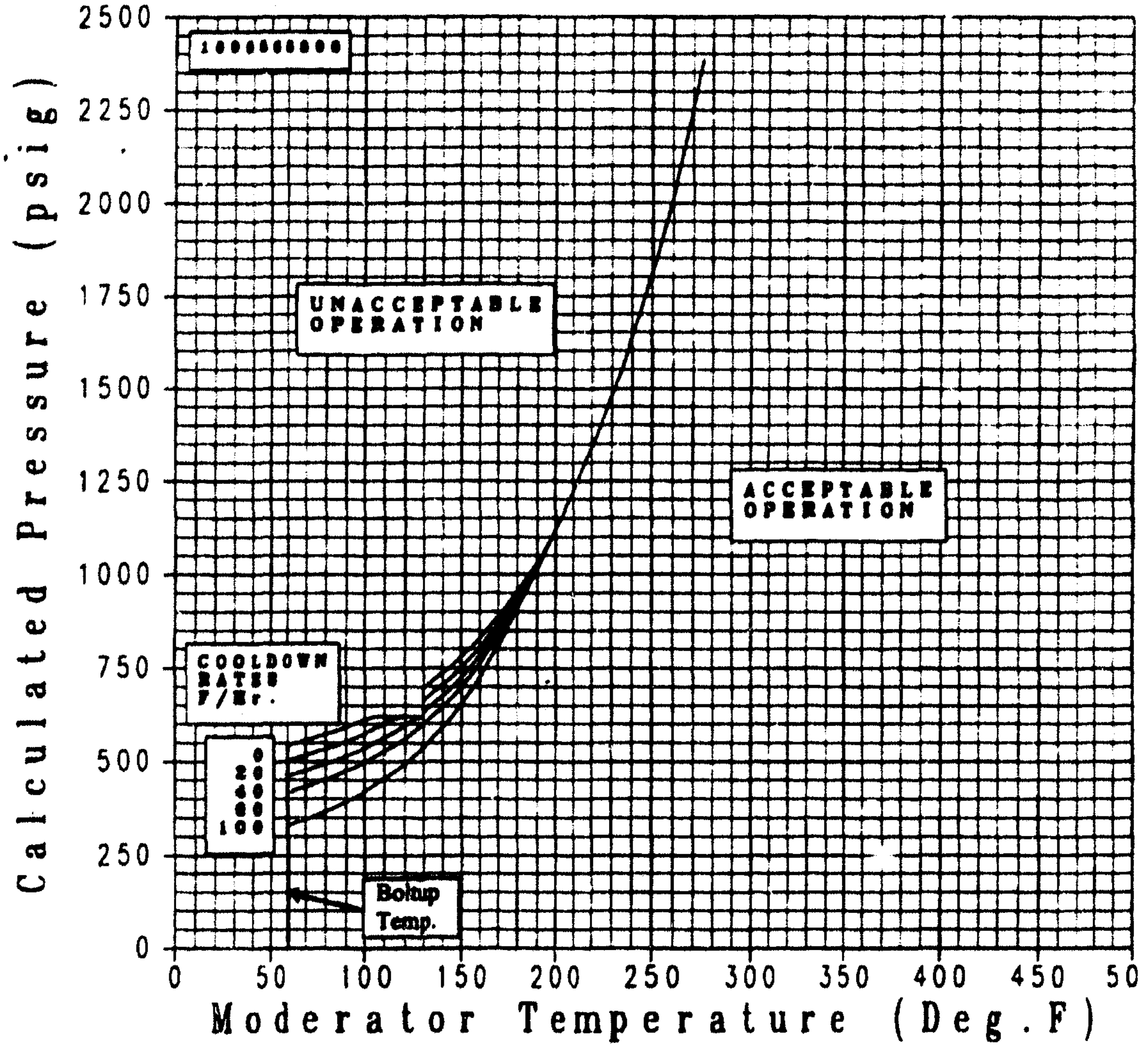

Figure 2.5.3-5
Cooldown Curves Applicable to 32 EFPY w/TPBAR
(without Uncertainties for Instrumentation Errors)

2.5.4 RESIDUAL HEAT REMOVAL SYSTEM (SRP 5.4.7) EVALUATION

The Residual Heat Removal System is used to remove decay heat and reactor coolant pump heat to cool the RCS from 350°F to either 200°F (cold shutdown) or 140°F (refueling shutdown). The ability of the RHRS to cool the RCS within applicable time limits depends partly upon the ability of the component cooling water system to transfer the heat from the RHRS and other auxiliary systems, such as the spent fuel pool cooling system, to its ultimate heat sink.

Generally, decay heat from the RCS will be reduced with TPBARs, due to the reduced average maximum burnup of the core. This reduced heat load on the RHRS, and therefore on the component cooling water system, will be offset to some degree by the increased heat load from the spent fuel pool heat exchangers, as described in Section 2.9.3. The net effect on the ability of the RHRS to cool the RCS within 20 hours for two-train (normal) cooldown and within 36 hours for single-train cooldown must be determined on a plant-specific basis.

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2.6 ENGINEERED SAFETY FEATURES

2.6.1 INTRODUCTION

The sections in Chapter 6 of the SRP deal with the engineered safety features (ESF) systems, which are designed to protect the public in the event of an accidental release of radioactive fission products from the RCS. Evaluations of the impact of TPBARs on mass and energy releases due to LOCAs and secondary system pipe ruptures, minimum containment pressure following a LOCA, and combustible gas control are described in Sections 2.6.2 through 2.6.5.

SRP 6.1.1
Engineered Safety Features Materials: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a; 10 CFR 50, Appendix A, GDC 1, 4, 14, 31, 35 and 41; and 10 CFR 50, Appendix B. They specifically deal with the compatibility of the materials and fluids of the Engineered Safety Features (ESF). Specific criteria relate to: fabrication and welding; maintaining appropriate chemistry and pH (minimum of 7.0) in the containment spray and core cooling water; the hydrogen generation rate due to corrosion of metals by containment spray; limitations on the use of nonmetallic thermal insulation; and requirements placed on protective coatings applied to ferritic steels, aluminum, stainless steel, zinc-coated concrete, or masonry surfaces. The effects of the TPC on reactor coolant chemistry were evaluated (see Section 3.8.3 for details). In the worst case for normal operation, where two TPBARs are breached, the reactor coolant chemistry remains well within the design limits for impurities and particulates. Given that the reactor coolant chemistry is not significantly affected, and the materials have not changed, there is no impact due to the TPC on this SRP.

SRP 6.1.2
Protective Coating Systems (Paints) - Organic Materials: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix B as it relates to the quality assurance requirements for the design, fabrication and construction of safety-related structures, systems and components. They specifically deal with the suitability and stability of protective coatings and other organics (e.g., cable insulation) under design basis accident (DBA) conditions. Since no new coatings or organic materials are being introduced in conjunction with the incorporation of the TPBARs, and since an evaluation has determined that the post-DBA containment environment (see Section 2.3.7) is not significantly impacted by the incorporation of the TPBARs, there is no impact in this area.

SRP 6.2.1
Containment Functional Design: This section is a composite of Sections 6.2.1.1 through 6.2.1.5. The acceptance criteria for each section is discussed separately below.

SRP 6.2.1.1.A
PWR Dry Containments, Including Subatmospheric Containments: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 13, 16, 38, 50 and 64. They specifically deal with the containment response analyses relative to the calculated containment design temperature and pressure, the containment depressurization time, the maximum external pressure caused by inadvertent operation of

containment heat removal systems, and the adequacy of the instrumentation provided to monitor the post-accident environment. The incorporation of the TPBARs does not impact the mass and energy releases for LOCA or secondary system pipe ruptures (see Section 2.6.2 and 2.6.3 below), therefore, the containment response is not impacted.

SRP 6.2.1.1.B
Ice Condenser Containments: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 13, 16, 38, 39, 40, 50 and 64. They specifically deal with the containment response analyses relative to the calculated containment design temperature and pressure, the maximum external pressure caused by inadvertent operation of engineered safety features, the ice condenser system and return fan system designs and surveillances, the maximum allowable operating deck steam bypass area, and the adequacy of the instrumentation provided to monitor the post-accident environment. The reference plant does not have an ice condenser containment. For an ice condenser plant, if the mass and energy releases do not change with the incorporation of TPBARs, there will be no impact on the containment response.

SRP 6.2.1.2
Subcompartment Analysis: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 4 and 50. They specifically deal with the adequacy of the analysis used to determine the differential pressure values for containment subcompartments. Specific criteria relate to whether the initial atmospheric condition assumptions are sufficiently conservative, the adequacy of the nodalization scheme, and the appropriate modeling of the vent flow behavior. Based on the conclusions in Section 2.6.2 that the LOCA mass and energy releases are not impacted by the incorporation of TPBARs, the subcompartments analysis is not impacted.

SRP 6.2.1.3
Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 50 and 10 CFR 50, Appendix K. They specifically deal with the adequacy of the determination of the mass and energy release to containment due to a LOCA, including the sources of stored and generated energy to be considered, the break size and location, and the conservatism that should exist in the analysis for each phase of the accident. An evaluation of the impact of the TPBARs on the LOCA mass and energy release was performed and is summarized in Section 2.6.2 of this document.

SRP 6.2.1.4
Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 50. They specifically deal with the adequacy of the determination of the mass and energy release to containment due to a steam or feedwater line break, including the sources of energy to be considered, the break size and location, and the conservatism that should exist in the analysis. An evaluation of the impact of the TPBARs on the secondary side pipe rupture mass and energy release was performed and is summarized in Section 2.6.3 of this document.

SRP 6.2.1.5

Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.46 and 10 CFR 50, Appendix K, paragraph I.D.2. They specifically deal with the conservatism required in the calculation of a minimum containment pressure following a LOCA. An evaluation of the impact of the TPBARs on the minimum containment pressure was performed and is summarized in Section 2.6.4 of this document.

SRP 6.2.2

Containment Heat Removal Systems: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 38, 39 and 40. They deal with the adequacy of the systems provided to remove heat from the containment under post-accident conditions. Specific criteria address 1) the requirement that the systems be designed to accommodate a single active failure; 2) the requirement that the net positive suction head (NPSH) available to the pumps in both the injection and recirculation phases of operation should be greater than the required NPSH, and that certain conservative assumptions be used in the analyses; 3) the requirements pertaining to the spray header and nozzle arrangements of the containment spray system, the nozzle atomizing capability, and the heat removal effectiveness of the spray droplets; 4) the heat removal capability of the fan coolers under post-accident environmental conditions; 5) the inclusion of provisions to prevent surface fouling or the accounting for surface fouling in establishing the heat removal capability of the heat exchangers; 6) adequate design consideration of sump hydraulic performance, evaluation of potential debris generation and associated effects including debris screen blockage, and RHR and CSS pump performance under postulated post-LOCA conditions; 7) the provision for periodic inspection and operability testing of the containment heat removal systems and system components; and 8) the requirement for instrumentation to monitor containment heat removal system and system component performance under normal and accident conditions. Based on the conclusions in Sections 2.6.2 and 2.6.3 that mass and energy releases for LOCA and secondary side pipe rupture do not increase due to incorporation of TPBARs, the containment heat removal system will not be impacted.

SRP 6.2.3

Secondary Containment Functional Design: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 4, 16 and 43; and 10 CFR 50, Appendix J. They specifically deal with the capability of the secondary containment system on dual containment designs to control the atmosphere within the secondary containment and contiguous areas. The reference plant does not have a secondary containment. For a plant with a secondary containment, there would be no impact as long as the mass and energy releases do not increase.

SRP 6.2.4

Containment Isolation System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 1, 2, 4, 16, 54, 55, 56 and 57, and 10 CFR 50 Appendix K. They deal with the adequacy of the provisions for containment isolation following a postulated accident. The criteria related to the number and location of

isolation valves; the valve positions for various conditions; the valve actuation signals; the actuation and control features; the mechanical redundancy of the valves; the acceptability of closed piping systems; the protection from missiles, pipe whip and earthquake; the design criteria applied to isolation barriers and piping; the detection of a need to isolate remote-manual-controlled systems; and the provisions for testing of the barriers are not impacted by the presence of the TPBARs. The selection of closure times of isolation valves is based on minimizing the release of containment atmosphere to the environs, to mitigate the offsite radiological consequences, and assure that ECCS effectiveness is not degraded by a reduction in the containment backpressure. Additional tritium in the accident source term does not result in any change in isolation philosophy. Tritium is a minor concern compared to the core activity that is assumed to be released to containment. Therefore, the containment isolation system is not impacted by TPBARs.

SRP 6.2.5
Combustible Gas Control in Containment: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.44; 10 CFR 50.46; and 10 CFR 50, Appendix A, GDC 5, 41, 42 and 43. They specifically deal with the analysis of hydrogen and oxygen production following a LOCA and the capability to monitor, mix, and reduce the combustible gas concentrations in containment. An evaluation of the impact of the TPBARs on the combustible gas concentrations and on the requirements for the combustible gas control systems, was performed and is summarized in Section 2.6.5 of this document.

SRP 6.2.6
Containment Leakage Testing: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 52, 53 and 54; and 10 CFR 50, Appendix J. They deal with the preoperational and periodic leak testing of the reactor containment, and of systems and components which penetrate the containment. Specifically, the leakage rate testing program must meet the requirements in Appendix J which defines the methods and frequency, as well as the test acceptance criteria, for Type A, B, and C leakage rate tests. The TPBARs do not impact the containment penetrations or isolation valves, and therefore, do not impact the capability to perform the leakage tests as prescribed in Appendix J.

SRP 6.2.7
Fracture Prevention of Containment Pressure Boundary: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 1, 16 and 51. They deal with the fracture toughness of the ferritic materials of components which are part of the containment pressure boundary. The presence of the TPBARs in the core does not impact the metallurgical properties of these components.

SRP 6.3
Emergency Core Cooling System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.46; 10 CFR 50, Appendix A, GDC 2, 4, 5, 17, 27, 35, 36 and 37; and 10 CFR 50, Appendix K. They specifically deal with the impact of postulated single failures on the operability of the ECCS; the availability of alternate sources of electric power; the provision for periodic inservice inspection of important components and operability testing of the system; the automatic actuation of the system by signals of suitable

diversity and redundancy; the provision for manual actuation, monitoring and control from the reactor control room; the prevention of damaging water hammer; the prevention of adverse effects on the system from nonsafety-related portions of the system; and the design of interfaces with other systems. The incorporation of the TPBARs does not require a design change for the ECCS based on the accident analysis results (Section 2.15). The only potential interaction of the TPBARs with the ECCS is in relation to the post-LOCA environment, and its impact on the equipment qualification. The evaluation in Section 2.3.7 concluded that current equipment qualification bounds the expected conditions with the TPBARs.

SRP 6.4
Control Room Habitability Systems: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 4, 5 and 19. They deal with the definition of areas to be included in the control room emergency zone, the effectiveness of the ventilation system relative to a single failure and the use of low leakage dampers or valves, the ability of the ventilation systems to pressurize the control room, the location of the control room inlets with respect to the potential release points, the toxic gas hazard limits, and the radiation hazard limits. Of these, only the radiation hazard limit criteria is potentially impacted by the incorporation of the TPBARs. An evaluation of the design basis LOCA source terms and the impact on the DBA doses to control room personnel was performed and is discussed in Section 2.15.6 of this document.

SRP 6.5.1
ESF Atmosphere Cleanup Systems: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 19, 41, 42, 43, 61 and 64. They deal with the capability of the ESF atmosphere cleanup systems to operate and retain radioactive material after a DBA; the provisions for air prefiltering, moisture removal, and charcoal adsorption; redundancy requirements; seismic requirements; automatic actuation; air flow rate limitation; minimum instrumentation requirements; component design, construction and testing requirements; and provisions for inplace testing. Operation with TPBARs has no impact on the design requirements or operability of the ESF filter systems since there are only minor changes to the source term for filterable activity.

SRP 6.5.2
Containment Spray as a Fission Product Cleanup System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 41, 42 and 43. They deal with the effectiveness of the containment spray and spray additive or pH control systems in removing fission products (iodine, in particular), when accident analyses rely on fission product scrubbing of the containment atmosphere for the mitigation of radiological consequences following a postulated accident. Specific criteria relate to 1) the automatic initiation of the spray system and automatic transfer from injection mode to recirculation mode, with a minimum operating period of at least two hours; 2) location of the spray nozzles to achieve maximum coverage; 3) the effectiveness of post-accident containment atmosphere mixing; 4) the design of the spray nozzles to minimize clogging; 5) the chemistry of the spray solution and its effectiveness for iodine absorption and retention; 6) provisions for containment sump solution mixing; 7) maintaining the pH level of the aqueous solution collected in the containment sump after injection of containment

spray and ECCS water to minimum level of 7 to assure long-term iodine retention; 8) the appropriate storage facilities for required additives; 9) the ability of the system to function with a single failure; 10) provisions for testing; 11) specification of appropriate limiting conditions in the plant Technical Specifications; and 12) the definition of acceptable methods for computing fission product removal rates. The physical design of the system is not being modified. The Spray Additive System is typically part of the Containment Spray System. Sodium hydroxide is added to the spray flow to raise the pH of the borated sump solution following a LOCA, in order to ensure the long-term integrity of stainless steel components. The RCS liquid is a small fraction of the total sump fluid, so minor changes in RCS boron concentration do not significantly affect the sump concentration, and therefore the sodium hydroxide required to be added. Although the Spray Additive System for the reference plant is not in Westinghouse scope, a review of the core boron requirements for TPBAR showed that there is little, if any, effect on the typical RCS boron concentration, and therefore, the Spray Additive System will not be affected by TPBARs. Credit for spray removal of airborne activity is only assumed for iodines, and this is not affected by the presence of TPBARs.

SRP 6.5.3
Fission Product Control Systems and Structures: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 41, 42 and 43. They deal with the acceptability of the model developed for estimating the radiological consequences of a design basis LOCA. Specifically, the minimum primary containment design leakage rates assumed should not be less than 0.1% per day, and the containment isolation methods and times must result in calculated doses within the guidelines of 10 CFR Part 100. The incorporation of the TPBARs does not change the leak rate assumed in the model used for dose calculations. The calculated doses for each DBA are discussed in Section 2.15.6 of this document.

SRP 6.5.4
Ice Condenser as a Fission Product Cleanup System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 41, 42 and 43. They deal with the effectiveness of the ice condenser system in removing elemental iodine (i.e., pH of post-accident recirculating solution is above 7) and the specification of appropriate limiting conditions in the Technical Specifications for testing, operation, and inspection of the system. The reference plant is not an ice condenser plant, however, it is concluded that incorporation of the TPBARs in an ice condenser plant will not impact the ability of the ice condenser to remove airborne iodines.

SRP 6.6
Inservice Inspection of Class 2 and 3 Components: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a and 10 CFR 50, Appendix A, GDC 36, 37, 39, 40, 42, 43, 45 and 46. They deal with the inservice inspection (ISI) program for the Class 2 and 3 components. Specific criteria relate to the lists of affected components subject to inspection, the accessibility of the components, the definition of the examination categories and methods, the appropriate inspection interval, the methods to be used in evaluating the results, the specifications for system pressure tests, augmented ISI for high-energy fluid system piping between containment isolation valves, any code exemptions,

and relief requests. None of the Class 2 and 3 components are being changed as a result of incorporating TPBARs, therefore the ISI is not impacted.

2.6.2 LOCA MASS AND ENERGY RELEASE (SRP 6.2.1.3) EVALUATION

2.6.2.1 Introduction

The mass and energy releases in LOCA transients must be considered for short term subcompartment analysis and long term containment integrity analysis. The releases are sensitive to different parameters depending upon the analysis in question. The short term subcompartment analysis is concerned with the maximum differential pressure that can occur across a compartment wall, floor, ceiling, or a major component support during the initial seconds following the rupture of a pipe. The long term containment integrity analysis focuses on the performance of the containment heat removal systems by assuring that the global containment pressure and temperature during a LOCA transient remain within the design limits and equipment qualification profiles. In general, bounding values for RCS conditions are opposite for these analyses. The short term releases are more conservative because of density effects when the RCS temperatures are biased low and the long term releases are more conservative due to enthalpy when the conditions are biased high.

2.6.2.2 Input and Methodology

The evaluation method involves a comparison of key safety analysis parameters used in the typical Westinghouse LOCA mass and energy analysis process for a 4 loop plant design for the short term and the long term transients. Following a qualitative assessment of these key parameters, a comparison is made with the respect to the analyses of record being evaluated. The key parameters for both the short term and the long term LOCA releases are the RCS conditions and the core stored energy for the long term analysis only. Since the RCS conditions in Table 1-1 for the TPC are identical to the conditions for the reference plant, it is concluded that there would not be any change in the reference plant short or long term LOCA mass and energy releases. An evaluation of the effect of a full complement of TPBARs showed a negligible impact on the core stored energy. The current methodology for core stored energy is sufficiently conservative to cover the relatively small amount of heat generated by the TPBARs. Section 2.3.6 of this report provides a comparison of core and vessel pressure drops with and without the TPBARs. The results show that the core and vessel pressure drops do change slightly (i.e. less than 3%) for normal operation. The pressure drop changes would not have an impact on the short term mass and energy release rates and would not be expected to have an appreciable impact on the long term mass and energy release rates as the system blows down or refills during a LOCA. In addition, Section 2.6.4 of this report provides information for the minimum containment backpressure calculations for determining the LOCA peak clad temperature. This evaluation determined that the presence of TPBARs could actually result in a decrease in the initial RCS inventory. A decrease in the inventory would be a benefit for the LOCA mass and energy releases for containment integrity analysis.

2.6.2.3 Acceptance Criteria and Results

There are no specific limits on the mass and energy releases for the short term and long term LOCA mass and energy releases for containment analyses. There are requirements that are described in SRP 6.2.1 and subsequent sub-sections. SRP 6.2.1.1.A and 6.2.1.1.B provide the requirements for dry containment, sub-atmospheric containment, and ice condenser containment designs. SRP 6.2.1.2 provides the necessary information for subcompartment analysis. The requirements for LOCA mass and energy releases for the containment analyses are described in SRP 6.2.1.3. Limiting criteria are specified for the results of the containment pressure and temperature response and subcompartment maximum differential pressures (for a LOCA inside containment) and the equipment environmental qualification (for a LOCA inside containment) analyses which use the mass and energy releases as input to the calculations. Since none of the key safety analysis parameters have changed for a core reload design with TPBARs, the licensing-basis analyses of record for the reference plant for LOCA mass and energy releases remain valid. Therefore, the containment analysis results with respect to mass and energy releases and the associated pressure and/or temperature response analyses also remain valid.

2.6.2.4 Conclusions

Based on the conclusion that none of the key safety analysis parameters are adversely impacted for a core reload design with TPBARs, the licensing-basis analyses of record for the short term and long term LOCA mass and energy releases continue to remain valid. The conclusions presented in the FSAR for the reference plant with respect to mass and energy releases and the associated pressure and/or temperature response analyses also remain valid. A plant specific evaluation is needed if RCS pressures and temperatures are outside a plant's operating window for their current design basis analyses or if the initial system inventory increased.

2.6.3 STEAMLINE AND FEEDLINE BREAK MASS AND ENERGY RELEASE (SRP 6.2.1.4) EVALUATION

2.6.3.1 Introduction

Mass and energy releases in non-LOCA accidents due to high-energy secondary-side line breaks (steamline and feedline breaks) are sensitive to changes in core reactivity coefficients; specifically, moderator density coefficients and shutdown margin. In general, bounding values of reactivity coefficients are used in the analyses to bound the accident over a wide range of core conditions.

2.6.3.2 Input and Methodology

The evaluation method involves a comparison of key safety analysis parameters used in the standard Westinghouse reload safety evaluation process (Reference 1). Following a qualitative assessment of these key parameters, a comparison is made with the respect to the high-energy secondary-side line break analyses of record being evaluated. As documented in Section 2.4.3, all of the key safety analysis parameters which comprise the reference plant safety analyses (for steamline break and feedline break mass & energy releases) bound the core reload design

values with TPBARs. In addition, the effects of the fuel temperatures associated with ZIRLO™ fuel rod cladding have been evaluated. In general, small changes in the fuel temperatures have little or no effect on non-LOCA safety analyses using the LOFTRAN code. In particular, secondary-side line break transients which calculate mass and energy releases are dominated by assumptions related to the main steam system design and protection and are not sensitive to core-related inputs such as fuel temperatures.

2.6.3.3 Acceptance Criteria and Results

There are no specific limits on the mass and energy releases calculated following a high-energy secondary-side line break. Limiting criteria are specified for the results of the containment pressure and temperature response (for a high-energy secondary-side line break inside containment) and the equipment environmental qualification (for a high-energy secondary-side line break outside containment) analyses which use the mass and energy releases as input to the calculations. However, the high-energy secondary-side line break safety analyses meet all the conditions with respect to sources of energy and calculations delineated in SRP 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures."

Since none of the key safety analysis parameters has been exceeded for the core reload design with TPBARs, the licensing-basis analyses of record for the high-energy secondary-side line breaks remain valid. Therefore, the analysis results with respect to mass and energy releases and the associated pressure and/or temperature response analyses also remain valid.

2.6.3.4 Conclusions

Based on the conclusion that none of the key safety analysis parameters has been exceeded for the core reload design with TPBARs, the licensing-basis analyses of record for the high-energy secondary-side line breaks continue to remain valid. The conclusions presented in the FSAR for the reference plant with respect to mass and energy releases and the associated pressure and/or temperature response analyses also remain valid.

2.6.3.5 References

1. WCAP-9273-NP-A (Non-Proprietary), "Westinghouse Reload Safety Evaluation Methodology", S.L. Davidson (Ed.), et al., July 1985.

2.6.4 MINIMUM CONTAINMENT PRESSURE (SRP 6.2.1.5) EVALUATION

An evaluation has been made of the effect of a full core of TPBARs on containment minimum pressure used in LBLOCA analysis. Installation of the TPBARs and the resulting decrease in primary system liquid volume results (conservatively) in a reduction in total mass and energy release to containment of approximately 1500 lbs and 550,000 Btu, respectively. These values are less than 0.2% of the total releases and are not considered significant with respect to LBLOCA peak cladding temperature or other LOCA limits.

2.6.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT (SRP 6.2.5) EVALUATION

2.6.5.1 Introduction

As noted above, the acceptance criteria for the design of the systems provided for combustible gas control are the relevant requirements of 10 CFR Part 50, Paragraphs 50.44 and 50.46 and GDC 5, 41, 42, and 43. As part of this acceptance criteria, analyses should indicate that a single system train is capable of maintaining the combustible gas concentrations to levels such that uncontrolled hydrogen/oxygen recombination would not take place. An assessment of the impact of the TPC core on post-LOCA hydrogen generation inside containment indicates that the TPC is not expected to have a significant impact on the total post-LOCA hydrogen produced in a plant that has operated with a conventional core design.

2.6.5.2 Evaluation

The TPC can impact the post-LOCA hydrogen generation inside containment by adding tritium and hydrogen to the hydrogen inventory that is generated from other sources. The sources that are considered to generate hydrogen following a LOCA in plants operating with conventional cores are:

a. metal-water reaction with the fuel cladding,

b. corrosion of materials,

c. radiolysis in the sump and core solutions, and

d. RCS inventory prior to the accident.

There are two potentially significant sources of additional post-LOCA hydrogen inventory associated with a TPC; they are:

1. metal-water reaction with the zirconium components associated with the TPBARs, and

2. tritium and hydrogen that exist in the TPBARs prior to the accident.

Although radiolysis, which is a function of decay energy of the fission products, could be marginally impacted by the TPC, the impact is considered to be negligible since the TPC has lower burnups than conventional cores. The TPC core average (equilibrium cycle) burnup is only 25,976 MWD/MTU, as compared to the reference plant cycle 8 core average burnup of

34,160 MWD/MTU. Also, corrosion of materials is a function of the post-LOCA temperature inside the containment. However, review of this area (Section 2.6.2) indicates that the associated changes would be negligible.

The first potential source of hydrogen identified above that can be attributed to the TPC design is that associated with the additional inventory of zirconium in the getter materials contained inside the TPBARs. Based on the chemical stoichiometry of the zirconium-water reaction (i.e., one pound mole of zirconium metal reacted must produce two pound moles of hydrogen), 7.9 standard cubic feet (scf) of hydrogen gas is produced for each pound of zirconium metal reacted. Then if the total mass of zirconium associated with the 3,344 TPBARs in an equilibrium cycle is converted to hydrogen, the amount of hydrogen gas is 7,600 scf.

It should be noted that the zirconium that is subject to the zirconium-water reaction is specified in 10 CFR 50.44 (Reference 1) to be only that associated with the ".. fuel cladding surrounding the active fuel region..." and "... the mass of metal in the cladding cylinders surrounding the fuel ..". This follows since it is generally only the metal in the active core region that is subject to the high temperatures, in excess of 1800°F, that are necessary for the zirconium-water reaction to occur.

However, the large break LOCA analysis for the TPC core (see Section 2.15.5.2) indicates that a fraction of the TPBARs could exceed 1800°F and would be expected to burst. Following expulsion of the gases, some diffusion of steam into the TPBAR could be postulated. For conservatism, the TPBAR internal zirconium components are treated in an analogous fashion to the treatment of the internal surface of fuel rod cladding following clad burst. For a fuel rod, zirconium oxidation is calculated on the internal surface over the length of the burst node. For the TPBAR, complete oxidation of the zirconium within the 3 inch long burst node is assumed with the resulting hydrogen added to the hydrogen calculated for the fuel. The fraction of the total absorber length represented by the burst node length is

$$F = 3 \text{ in} / 128.5 \text{ in} = 0.0234,$$

which is also the fraction of the total hydrogen inventory that could be released based on a zirc-water reaction with all of the TPBAR zirconium mass. Then, the equivalent hydrogen that could be released is

$$V' = 7{,}600 \times 0.0234 = 178 \text{ scf}.$$

In addition to the contribution from the zirc-water reaction, another potential source of hydrogen and tritium is that contained within the TPBARs. Although inconsistent with criteria as specified in 10CFR50.46 (Reference 3) (i.e. that the calculated fraction of the cladding in the core region that is subject to the zirc-water reaction be less than 1%), USNRC Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident (Reference 4), specifies that a core melt scenario be considered in defining fission product sources. These sources are used in defining the hydrogen generation from core and sump solution radiolysis. For conservatism, a "core melt" scenario was considered in the release of tritium from the TPBAR getter material, i.e. all of the tritium was assumed to be released to the containment. As indicated in Section 2.4.3, the amount of tritium produced in

the TPC core designs are 2860 and 2805 grams of tritium for the initial and equilibrium cycles, respectively. Conservatively assuming 0.9 grams/rod (3000 grams for 3344 rods) and converting this amount of tritium (T) to the equivalent volume of tritium gas (T_2), results in a volume of 397 scf of T_2. An additional source of hydrogen associated with the TPBARs is that generated from the $^3He(n,p)T$ reaction inside the rods. At end of life, this source could generate an additional 9 scf which is available for release following a LOCA.

2.6.5.3 Conclusions

The total amount of the additional tritium and hydrogen that is conservatively estimated to be associated with a TPC is 584 scf. This inventory would be expected to exist in the primary coolant as water or tritiated water (HTO or T_2O), rather than as a gas. However, even if the complete hydrogen/tritium inventory associated with a TPC is conservatively assumed to be released to the containment atmosphere as gas, the added inventory represents only a 8.3% increase in the amount of hydrogen gas that is released immediately after a LOCA for the reference plant operating with a conventional core. The reference plant inventory with a conventional core is based on a zirconium-water reaction of 1.5% of the core cladding involved in the reaction (15.15 lb-moles or 5440 scf) and an RCS hydrogen inventory of 1660 prior to the accident (Reference 2).

Note that the other time-dependent sources of post-LOCA hydrogen, that is, the hydrogen from corrosion of materials and radiolysis in the sump and core solutions, are not considered, such that the fractional increase will reduce with time after LOCA as these other time-dependent sources add additional hydrogen inventory to containment. Also note that the fractional increase would be smaller than 8.3% for plants with higher zirconium-water reaction fractions. The amount of zirconium involved in the zirconium-water reaction is mandated by 10 CFR 50.44 to be 5 times the fraction calculated in the 10 CFR 50.46 ECCS performance criteria assessment (Reference 3). A value of 5% is an upper limit on this fraction since 10 CFR 50.46 specifies that the calculated fraction cannot exceed 1% of the cladding in the active core region and 5% is 5 times this limiting value. As noted above, the value associated with the reference plant is only 1.5%.

The lower flammability limit for hydrogen in the containment atmosphere that should not be exceeded (Reference 4 - USNRC Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-Of-Coolant Accident") is 4 volume percent. For the reference plant with a total containment free volume of 2,750,000 ft^3, a concentration of 4 volume percent equates to roughly 100,000 scf of hydrogen. Thus, the contribution of the TPC tritium inventory to the recommended Regulatory Guide limit is only

$$F' = 584/100{,}000 = .0058,$$

or less than 1% of the total from all sources. Further, even for a plant with a much smaller containment volume (i.e., an ice condenser plant), the projected fraction of total is only about 1%, in that the ice condenser plant containment volume is about half that associated with a plant with a dry containment. Thus, it is concluded that even based on highly conservative assumptions, the contribution of hydrogen from TPBARs does not exceed 1% of the limiting post-LOCA hydrogen inventory inside the containment.

2.6.5.3 References

1. USNRC Code of Federal Regulations, 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors".

2. Reference Plant, Final Safety Analysis Report Update, Section 6.2.5.3.

3. USNRC Code of Federal Regulations, 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors".

4. USNRC Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-Of-Coolant Accident", Revision 2, November 1978.

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2.7 INSTRUMENTATION AND CONTROLS

2.7.1 INTRODUCTION

The sections in Chapter 7 of the SRP deal with various plant instrumentation and control systems; the primary purposes of which are to provide automatic protection and exercise proper control against unsafe and improper reactor operation during steady-state and transient power operations, and to provide initiating signals to mitigate the consequences of faulted conditions. Evaluations of the impact of the TPBARs on the reactor trip system and engineered safety features actuation system, the systems required for safe shutdown and information systems important to safety, and control systems are described in Sections 2.7.2 through 2.7.4 of this document.

SRP 7.1
Instrumentation and Controls - Introduction: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 1. They deal with the proper identification of the instrumentation and control systems important to safety along with the identification of the acceptance criteria and guidelines (e.g., GDCs, IEEE Standards, Regulatory Guides, and Branch Technical Positions) applicable to each system. The incorporation of the TPBARs does not change the safety classification of the instrumentation and control systems, or the applicable acceptance criteria and guidelines.

SRP 7.2
Reactor Trip System: The acceptance criteria in this section deal with the capability of the reactor trip system (RTS) to initiate reactor shutdown to assure that specified acceptable fuel design limits are not exceeded. The acceptance criteria are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4, 13, 19, 20, 21, 22, 23, 24, 25, and 29; and 10 CFR 50.55a (h) with regards to IEEE Standard 279. Specifically, they relate to: 1) the identification of systems and components for the RTS which are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles; 2) the capability of the system to sense accident conditions and anticipated operational occurrences to initiate reactor shutdown consistent with the accident analysis; 3) the adequacy of the system reliability and testability; 4) the system independence; 5) the design of the system to fail into a safe mode; 6) the requirements of IEEE Std 279 with respect to control and protection system interactions; 7) the capability of the system to assure that fuel design limits are not exceeded for malfunctions of the reactivity control system such as accidental withdrawal of control rods; and 8) the compatibility of the RTS design with the functional performance requirements of essential auxiliary support (EAS) systems. The incorporation of the TPBARs will not necessitate a change to the physical or logical layout of the RTS, therefore the capability of the system to meet the requirements relative to system reliability, testability, independence, failure mode, control and protection system interaction, and compatibility with EAS systems remains unchanged; as does the identification of systems and components designed to survive natural phenomena, missiles, and abnormal environments. An evaluation of the capability of the system to initiate reactor shutdown in response to conditions specified in the safety analyses (including reactivity control system malfunction) was performed and is summarized in Section 2.7.2 of this document.

SRP 7.3
Engineered Safety Features Systems: The acceptance criteria in this section deal with the capability of the engineered safety features actuation system (ESFAS) to initiate operation of necessary engineered safety features (ESF) systems to prevent or mitigate damage to the core and reactor coolant system components and to ensure containment integrity, and with the capability of the ESF control systems to regulate the operation of ESF systems following their initiation. The applicable acceptance criteria are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4, 13, 19, 20, 21, 22, 23, 24, and 29; and 10 CFR Part 50.55a (h) with regards to IEEE Standard 279. Additional criteria applying to the ESF control systems are GDC 34, 35, 38, and 41 with regards to conformance to the single failure criterion on a system basis, and to operability from onsite and offsite electrical power. Specific criteria relate to: 1) the identification of systems and components for the ESFAS and ESF control systems which are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles; 2) the capability of the ESFAS to sense accident conditions and anticipated operational occurrences to initiate operation of ESF and EAS systems consistent with the accident analysis; 3) the adequacy of the ESFAS reliability and testability; 4) the ESFAS system independence; 5) the design of the ESFAS to fail into a safe mode; 6) the requirements of IEEE Std 279 with respect to control and protection system interactions; and 7) the compatibility of the ESFAS and ESF control system design with the functional performance requirements of essential auxiliary support (EAS) systems. The incorporation of the TPBARs will not necessitate a change to the physical or logical layout of the ESFAS or ESF control system, therefore the capability of the systems to meet the applicable requirements relative to system reliability, testability, independence, failure mode, control and protection system interaction, and compatibility with EAS systems remains unchanged; as does the identification of systems and components designed to survive natural phenomena, missiles, and abnormal environments. An evaluation of the capability of the ESFAS to initiate operation of necessary ESF systems in response to conditions specified in the safety analyses was performed and is summarized in Section 2.7.2 of this document.

SRP 7.4
Systems Required for Safe Shutdown: The acceptance criteria in this section deal with the capability of certain instrumentation and control systems to achieve and maintain a safe shutdown condition of the plant. The required systems are those which permit the necessary operations that prevent the reactor from returning to criticality, and provide an adequate heat sink so that design and safety limits on reactor coolant system temperature and pressure are not exceeded. The applicable acceptance criteria are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4, 13, and 19. Additional criteria applying to the systems required for safe shutdown are GDC 34, 35, and 38 with regards to conformance to the single failure criterion, and to operability from onsite and offsite electrical power. Specific criteria relate to: 1) the identification of systems and components required for safe shutdown which are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles; 2) the provision of instrumentation and controls to maintain variables and systems which can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems within prescribed operating ranges during plant shutdown; 3) the provisions within the control room to allow actions to be taken to maintain a safe condition during shutdown including a shutdown following an accident; 4)

the provision of equipment at appropriate locations outside the control room to achieve prompt hot shutdown and maintain a safe condition during hot shutdown, and to subsequently achieve cold shutdown; and 5) the compatibility of the design of the systems required for safe shutdown with the functional performance requirements of essential auxiliary support (EAS) systems. An evaluation of the impact of the TPBARs on these systems is described in Section 2.7.3 of this document.

SRP 7.5
Information Systems Important to Safety: The acceptance criteria in this section deal with the adequacy of the plant instrumentation to provide information for manually initiated and manually controlled safety functions, to indicate that plant safety functions are being accomplished, and to provide information from which appropriate actions can be taken to mitigate the consequences of anticipated operational occurrences and accidents. The applicable acceptance criteria are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4, 13, and 19. Specific criteria relate to: 1) the identification of systems and components for the information systems which are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles; 2) the physical independence of electrical systems; 3) the capability to monitor appropriate variables with a range and accuracy consistent with the plant safety analysis; and 4) the provision of information in the control room from which actions can be taken to operate safely under normal conditions and maintain a safe condition under accident conditions. An evaluation of the impact of the TPBARs on these systems is described in Section 2.7.3 of this document.

SRP 7.6
Interlock Systems Important to Safety: The acceptance criteria in this section deal with the adequacy of interlock systems important to safety which operate to reduce the probability of occurrence of specific events or to maintain safety systems in a state to assure their availability in an accident. These include those interlock systems which prevent overpressurization of low pressure systems when connected to high pressure systems, prevent overpressure of the primary coolant system during low temperature operation, assure availability of ECCS accumulators, isolate safety systems from non-safety systems, and preclude inadvertent interaction between redundant or diverse safety systems. The applicable acceptance criteria are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4, 13, and 19. Specific criteria relate to the design of these systems consistent with the plant safety analysis, single failure criteria, redundancy, independence, qualification, and testability. The incorporation of the TPBARs will not necessitate a change to the design of the interlock systems important to safety and no new accidents have been identified, therefore the capability of the systems to meet the applicable requirements has not changed.

SRP 7.7
Control Systems: The acceptance criteria in this section deal with the adequacy of the control systems used for normal operation which control plant processes having a significant impact on plant safety, but which are not relied upon to perform safety functions following anticipated operational occurrences or accidents. The applicable acceptance criteria are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 13 and 19.

An evaluation of the capability of the control systems to accommodate operational transients with TPBARs incorporated has been performed and is summarized in Section 2.7.4 of this document.

2.7.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES SYSTEM (SRP 7.2 AND 7.3) EVALUATION

2.7.2.1 Introduction

The primary purpose of the Reactor Trip System (RTS) and Engineered Safety Features System is to initiate a plant shutdown via control rod insertion, based on the magnitude of monitored plant parameters. The corresponding function of the Engineered Safety Features Actuation System (ESFAS) is to initiate necessary safety system actuation and/or a plant shutdown. The nominal setpoints for the monitored plant parameters, which are reflected in a plant's Technical Specifications, are selected such that there is sufficient allowance between the nominal setpoint and the safety limit to account for uncertainties. The uncertainties to be accounted for include process effects as well as instrumentation uncertainties.

2.7.2.2 Evaluation

The RTS/ESFAS setpoints are impacted by: 1) the safety analysis limits (SAL), 2) the type of instrumentation (sensors and racks), 3) the environment (both normal and abnormal) to which the instrumentation will be exposed, 4) the surveillance interval, 5) the calibration procedures for the instrumentation, 6) the measurement and test equipment (M&TE) used to calibrate the instrumentation, and 7) the Process Measurement Accuracy (PMA) terms which account for non-instrument effects. Based on the safety evaluations/analyses (see Section 2.15), no SAL changes have been required. The TPC will not employ different instrumentation, calibration procedures, surveillance intervals, or M&TE. The instrumentation environment (normal and abnormal) remains within the equipment qualification envelope for the reference plant (see Section 2.3.7).

PMA terms which might be impacted by the introduction of TPBARs are the radial power redistribution terms in the Power Range, Intermediate Range and Source Range Neutron Flux trips (the NIS channel trips), the rod shadowing term in the Intermediate Range Flux trip, and the RCS hot leg streaming burndown effects in the Overtemperature Delta T (OTΔT) and Overpower Delta T (OPΔT) trips.

Based on an evaluation of the impact of the TPBARs on edge assembly powers, it has been concluded that the TPBARs should not exacerbate the temperature gradient across the hot leg. Therefore, the hot leg streaming burndown effects for the TPC should be bounded by those present in the reference core. The radial power redistribution terms in the NIS trips are traditionally conservative allowances based on engineering judgment. Based on the control rod patterns, and insertion and withdrawal sequences remaining the same, the qualitative arguments used to support the radial power redistribution allowances incorporated in the standard Westinghouse uncertainty calculations for the NIS trips should remain valid for the TPC. In addition, based on the similarity of the TPC axial power shape to that for a typical

plant, the rod shadowing effect on the Intermediate Range NIS is not anticipated to be impacted by the introduction of the TPBARs.

2.7.2.3 Conclusions

Based on the above qualitative discussion, it is anticipated that the introduction of a full complement of TPBARs into the reference plant, as described in Section 2.4.3, will not impact the RTS/ESFAS setpoints. However, since some of the pertinent parameters might be impacted by optimizations performed on a plant specific basis, these setpoints should be evaluated to support the license amendments for the selected plants.

2.7.3 SAFE SHUTDOWN SYSTEMS AND INFORMATION SYSTEMS IMPORTANT TO SAFETY (SRP 7.4 AND 7.5) EVALUATION

2.7.3.1 Introduction

The purpose of the Safe Shutdown Systems is to provide the instrumentation and controls necessary to achieve and maintain a safe shutdown of the plant. These systems maintain the reactor subcritical and provide a heat sink to maintain pressure and temperature within acceptable limits. The purpose of the Information Systems Important to Safety is to monitor the plant during normal operations, anticipated operational occurrences, and post accident conditions. This instrumentation monitors five classes of variables:

- Variables required to support preplanned manual operator actions in response to design basis events,
- Variables required to monitor the status of critical safety functions (subcriticality, RCS pressure, RCS inventory, core cooling, heat sink, and containment),
- Variables required to monitor for a breach of the barriers to fission product release,
- Variables required to assess the operation of safety systems, and
- Variables required to determine the magnitude of the release of radioactive materials.

The Information Systems Important to Safety also include the Bypassed and Inoperable Status Indication (BISI) System, which provides automatic system level indication when engineered safety features are placed in a bypassed or inoperable state.

2.7.3.2 Evaluation

The functional performance of the Safe Shutdown Systems and Information Systems Important to Safety is determined by a variety of factors including the system hardware and instrumentation design, the surveillance intervals, the calibration processes and equipment, the ranges of the process variables that are monitored and controlled, the environmental conditions to which the instrumentation is exposed, and any process measurement effects which may influence instrument accuracy.

The introduction of TPBARs will not change the design of the system hardware and instrumentation, nor will it affect the surveillance intervals or the calibration processes and equipment. The ranges of the process variables that are monitored and controlled are unaffected by the TPBARs since the operating parameters for the tritium producing core will be unchanged for the reference plant. That is, the process pressures, temperatures, power levels, and flows will be unchanged. The environmental conditions for the TPC have been determined to be bounding for the reference plant (Section 2.3.7). With respect to instrument accuracy, the process measurement terms that could affect the post accident monitoring uncertainties required by the Emergency Response Guidelines have been considered. With the exception of the core exit thermocouples, all monitored variables are external to the core and are unaffected by the presence of the TPBARs. For the conventional top mounted thermocouples (such as installed in the reference plant) which are located above the fuel assemblies in the upper support column, the bulk temperature of the core exit fluid will continue to be measured in the presence of the TPBARs and will, therefore, be unaffected. Several Westinghouse plants employ bottom mounted thermocouples (BMTCs), which are an integral part of the bottom mounted instrumentation which is located in the fuel assembly instrumentation guide tubes. In this location the thermocouples measure core bypass flow temperature, rather than the core exit stream, and are susceptible to large process errors, which have been quantified through a combination of modeling and testing. Although the installation of the TPBARs is not expected to affect the magnitude of the previously determined process errors, this should be evaluated on a plant specific basis if TPBARs are to be installed in a plant with BMTCs.

2.7.3.3 Conclusions

In summary, based on the above qualitative evaluation, it is anticipated that the introduction of TPBARs into the reference plant will not affect the Safe Shutdown Systems and Information Systems Important to Safety. However, for plants that employ BMTCs, the process measurement effects previously calculated for post accident monitoring should be revalidated for the presence of TPBARs.

2.7.4 OPERATIONAL TRANSIENTS/MARGIN TO TRIP (SRP 7.7) EVALUATION

2.7.4.1 Introduction

Nuclear power plants are required to be able to perform certain operational transients without challenging the reactor trip or engineered safeguards setpoints. These are primarily at-power load change transients induced by turbine generator load changes, and should be able to be controlled by the normally operating NSSS control systems (reactor rod control, pressurizer pressure control, pressurizer level control, steam generator level control, and steam dump control). This section will assess the TPC plant's capability to handle the normal operational transients acceptably.

2.7.4.2 Methodology

The normal operational transients that the plant should be able to handle include:

a. Normal loading and unloading at 5%/minute between 15% and 100% power

b. 10% step load increase or decrease between 15% and 100% power

c. Large load rejection with steam dump. This can be a maximum of either a 50% or 100% load decrease, depending upon the plant. For the representative plant being considered for TPBAR inclusion, this is a 50% load decrease.

d. Turbine trip without a reactor trip below the P-9 power level setpoint (50% of rated power).

Of all of the above transients, the one that is the most limiting is the large load rejection transient. This transient generally shows the lowest margin to the reactor trip setpoints and challenges all of the NSSS control systems. Therefore, it was decided to analyze this transient to determine the effect of the TPC.

The methodology used was to analyze a large load rejection for the reference core and the TPC.

The analysis was done using the LOFTRAN computer code (Reference 1). This code has been used by Westinghouse for approximately 20 years for the analysis of plant operational transients and determining the acceptability of NSSS control system behavior and response and plant margin to trip.

2.7.4.3 Significant Input Parameters and Assumptions

The following were the key input parameters and assumptions used for this evaluation:

1. NSSS performance parameters for the TPC (Table 1-1). These include the rated power level, RCS operating temperature and pressure, RCS flow, steam generator type, and steam generator secondary side steam pressure, steam/feedwater flow, and feedwater temperature.

2. NSSS performance parameters for the reference plant (see Table 1-1).

3. Fuel reactivity parameters for the TPC.

4. Fuel reactivity parameters for the reference core, based on the Nuclear Design Report.

5. It was assumed that the design basis operational behavior of the plant would not be revised as a result of the TPC. This is based upon the following assumptions:

    a. The TPBAR, inserted into the thimble, is designed to meet the requirements of a large representative four-loop Westinghouse reactor under Conditions I, II, III and IV, defined in the FSAR of the reference plant.

    b. The use of the TPBARs will not cause or create any new accident scenarios other than those currently identified in Regulatory Guide 1.70 and the NRC Standard Review Plan nor add to the probability of existing scenarios.

c. There will be no plant power uprating (or derating) or RCS thermal-hydraulic parameter changes made to enhance reactor performance or operational flexibility for the TPC.

The LOFTRAN input deck for the reference plant was used to represent the physical NSSS component geometry. For the purposes of the operability analyses discussed here, the inclusion of the TPBARs only affects the fuel reactivities. There are no differences in the plant rating, RCS flow, primary or secondary side pressures or temperatures for the TPC vs. the reference plant. There are no changes identified in the reactor trip or engineered safeguards actuation setpoints (see Section 2.7.2). There are no changes being specified in the steam dump operation. No control system setpoint revisions are being specified for the TPC; setpoint revisions will only be made if unacceptable results are obtained with the operational transient analyses. The only differences made in the analysis of the TPC vs. the reference reactor were in the fuel reactivities and control rod worths.

The analysis methodology used was similar to that used for generic margin to trip analyses:

- Plant initially at 100% power with all NSSS control systems in automatic
- Fuel reactivities considered are for beginning of core life (most restrictive values for margin to trip and transient stability)
- Rod are at all-rods-out point (most limiting value; give minimum rod worths and least amount of initial transient mitigation)
- Transient initiated by reducing the turbine load from 100 to 50% at 200%/minute (generic maximum unloading rate)

2.7.4.4 Results and Acceptance Criteria

The analysis of the large load rejection transient for both the TPC and the reference plant showed no significant difference in the response between the two plants. Acceptable margin to the reactor trip setpoints (Overpower and Overtemperature ΔT trips are limiting functions) was obtained, and the plant response was stable and non-oscillatory. There was no significant impact of the inclusion of the TPC on the response characteristic of the plant during the analyzed transient.

2.7.4.5 Conclusions and Affected Documents

Based upon the results obtained for the limiting transient analyzed, the TPC will not noticeably change the response characteristic of the plant from that seen for the reference plant. There are no control system setpoint revisions required due to the TPC.

The documents that are normally affected by the operational transients/margin to trip would include:

a. Control system setpoint study

b. Precautions, Limitations and Setpoints document

c. Relevant FSAR sections on control system operability (FSAR section 7.7 for reference plant)

Since there are no significant impacts of the TPC on the plant operational transient response, and no control system setpoint changes are required, none of the above potentially impacted documents would require revisions for the reference plant.

2.7.4.6 Potential Plant Specific Analysis

For other candidate plants, there should be no impact as long as the NSSS performance parameters and the protection system setpoints don't change, and the fuel reactivity changes are minimal.

2.7.4.7 References

1. WCAP-7907, "LOFTRAN Code Description", Non-Proprietary, Burnett, T. W. T., October 1972.

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2.8 ELECTRIC POWER

The sections in Chapter 8 of the SRP deal with the offsite and onsite power systems. Based on the following qualitative discussions, it was determined that evaluations of potential TPBAR impact were not required and it is not anticipated that there would be changes to Chapter 8 of a typical SAR as a result of incorporating TPBARs.

SRP 8.1
Electric Power - Introduction: The acceptance criteria in this section deal with the description of the utility grid and the interconnections between the nuclear unit, the utility grid, and other grids, along with the identification of the acceptance criteria and guidelines (e.g., GDCs, IEEE Standards, Regulatory Guides, and Branch Technical Positions) applicable to the design of electric power systems. The incorporation of the TPBARs does not change the grid connections, or the applicable acceptance criteria and guidelines.

SRP 8.2
Offsite Power System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 5, 17 and 18. They deal with 1) the sharing of circuits of the offsite power system between units; 2) the capacity and capability of the offsite power system to permit functioning of structures, systems, and components important to safety; 3) the provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or loss of onsite power; 4) the physical independence of circuits; 5) the availability of circuits; and 6) the testability of the system. The incorporation of the TPBARs does not impact the capacity or the physical and electrical layout of the offsite power system.

SRP 8.3.1
A-C Power Systems (Onsite): The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4, 5, 17, 18 and 50. They deal with the adequacy of the onsite a-c power system with regard to 1) the required redundancy; 2) the single failure criterion; 3) protection from the effects of postulated accidents; 4) testability; and 5) the capacity and capability to supply power to all safety loads and other required equipment. The incorporation of the TPBARs does not impact the capacity or the physical and electrical layout of the onsite a-c power system. In addition, the evaluation for SRP 3.11 (Section 2.3.7) determined that the incorporation of the TPBARs will not cause the post-accident environment to exceed the equipment qualification envelope.

SRP 8.3.2
D-C Power Systems (Onsite): The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4, 5, 17, 18 and 50. They deal with the adequacy of the onsite d-c power system with regard to 1) the required redundancy; 2) the single failure criterion; 3) protection from the effects of postulated accidents; 4) testability; and 5) the capacity and capability to supply d-c power to all safety loads and other required equipment. The incorporation of the TPBARs does not impact the capacity or the physical and electrical layout of the onsite d-c power system. In addition, the evaluation for SRP 3.11 (Section 2.3.7) determined that the incorporation of the TPBARs will not cause the post-accident environment to exceed the equipment qualification envelope.

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2.9 AUXILIARY SYSTEMS

2.9.1 INTRODUCTION

The sections in Chapter 9 of the SRP deal with the various auxiliary systems required in the plant. Evaluations of the impact of TPBARs on the new and spent fuel storage areas, the spent fuel pool cooling and cleanup system, the component cooling water system, the demineralized water makeup system, and the process and post-accident sampling system are described in Sections 2.9.2 through 2.9.5 of this document.

SRP 9.1.1
New Fuel Storage: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 5, 61 and 62. They deal with the storage capacity of the design, the capability of the storage racks (by their design and arrangement) to maintain an acceptable degree of subcriticality during all storage conditions, the effects of external loads and forces (e.g., safe shutdown earthquake, crane uplift forces) on the racks and vault, the effects of sharing with another unit, and the potential impact of failure of other plant equipment close to the new fuel storage facility. Specifically they require that 1) a k_{eff} not greater than 0.95 is maintained when the array is fully loaded and flooded with nonborated water; 2) a k_{eff} not greater than 0.98 is maintained with fuel of the highest anticipated reactivity in place assuming moderation by aqueous foam; 3) the racks are designed to prevent insertion of a fuel assembly anywhere other than in a design location and provisions have been made for drainage to prevent accumulation of a fluid moderator; 4) failures of nonsafety-related systems or structures not designed to seismic Category I in the vicinity of the storage facility will not cause k_{eff} to exceed the allowance; 5) storage racks and anchorages can withstand the maximum uplift forces from the lifting devices without an increase in k_{eff}; 6) the design precludes damage from dropped heavy objects; 7) the sharing of a storage facility does not result in the potential for increasing k_{eff}; and 8) the safety function of the facility will be maintained if subjected to adverse natural phenomena such as earthquakes, tornadoes, hurricanes, and floods. A review of these requirements shows that only the subcriticality and the seismic response are potentially impacted by TPBARs. An evaluation of the adequacy of the subcriticality of the rack arrangement, and the impact of the TPBAR assembly weight on the seismic response of the racks for the reference plant is summarized in Section 2.9.2 of this document.

SRP 9.1.2
Spent Fuel Storage: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4, 5, 61, 62 and 63. They deal with the adequacy of the storage capacity for spent fuel, the adequacy of the storage rack design and arrangement to maintain a subcritical condition, the effects of external loads and forces, and the design codes, materials compatibility, and shielding requirements. Specifically they require that 1) the storage capacity for a dual shared storage pool shall equal or exceed one full core discharge plus two normal fuel discharge cycles; 2) a k_{eff} not greater than 0.95 is maintained when the array is fully loaded and flooded with nonborated water; 3) the racks are designed to prevent insertion of a fuel assembly anywhere other than in a design location; 4) failures of nonsafety-related systems or structures not designed to seismic Category I in the vicinity of the storage facility will not cause k_{eff} to exceed the allowance; 5)

storage racks and anchorages can withstand the maximum fuel handling equipment uplift forces without an increase in k_{eff} or a decrease in pool water inventory; 6) the weight of all loads being handled above stored spent fuel shall not exceed that of one fuel assembly and its associated handling tool, and the potential energy of all lighter loads shall not exceed that of a single fuel assembly and its associated handling tool when dropped from its normal operating height; 7) the safety function of the facility will be maintained if subjected to adverse natural phenomena such as earthquakes, tornadoes, hurricanes, and floods; 8) the water level will be maintained to provide radiation shielding to achieve personnel exposures as low as reasonably achievable (ALARA), and 9) the fuel shipping cask loading pit has been designed so that the safety function of the integrated system will be maintained during adverse environmental conditions, the fuel transfer canal can be isolated from the fuel pool and the loading pit, and a dropped heavy load in the pit area will not result in an unacceptable loss of fuel pool water. Recently, a Technical Specification revision was approved for the reference plant which allows credit for soluble boron in the spent fuel pool for maintenance of subcriticality. This revision effectively replaces the subcriticality requirement above (item #2) with the following two criteria; 1) k_{eff} <1.0 when fully flooded with unborated water, and 2) $k_{eff} \leq 0.95$ when fully flooded with water borated to a specified concentration. A review of the requirements shows that only the subcriticality, the seismic response and personnel exposures are potentially impacted by TPBARs. An evaluation of the subcriticality and the effect on seismic characteristics of the spent fuel racks with the incorporation of the TPBARs is summarized in Section 2.9.2 of this document. ALARA evaluations are contained in Section 2.12.3.

SRP 9.1.3

Spent Fuel Pool Cooling and Cleanup System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 20.1(as it relates to ALARA) and 10 CFR 50, Appendix A, GDC 2, 4, 5, 44, 45, 46, 61 and 63. They deal with the design of the spent fuel pool cooling and cleanup system and its makeup system. Specifically, they relate to 1) the capability of the system and structures to withstand the effects of natural phenomena (e.g. earthquakes, tornadoes, and hurricanes) and external missiles; 2) the capability of shared systems and components important to safety to perform required safety functions; 3) the capability to transfer heat loads from safety-related structures, systems, and components to a heat sink under normal and accident conditions; 4) redundancy of components to assure performance of safety functions with a single active failure coincident with the loss of all offsite power; 5) the capability to isolate components, systems, or piping to prevent compromise of the system safety function; 6) design provisions to permit periodic inspection of safety-related components and equipment; 7) design provisions to permit operational functional testing of safety-related systems or components to assure structural integrity and system leak tightness, operability, and adequate performance of active system components, and the capability of the integrated system to perform required functions; 8) provisions for containment of radioactive materials; 9) provisions for decay heat removal; 10) capability to prevent reduction in fuel storage coolant inventory under accident conditions; 11) capability and capacity to remove corrosion products, radioactive materials and impurities from the pool water; and 12) provision of appropriate monitoring systems. The spent fuel pool cooling and cleanup system design is not being modified as a result of TPBAR incorporation, therefore there is no impact on the capability to withstand natural phenomena and external missiles, the redundancy of components, the capability to

isolate, the provisions for inspection and testing, the prevention of coolant inventory reduction under accident conditions, and the provision of monitoring systems. An evaluation of the spent fuel pool heatup with the incorporation of the TPBARs is summarized in Section 2.9.3 below. ALARA evaluations are contained in Section 2.12.3.

SRP 9.1.4
Light Load Handling System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 5, 61 and 62. They deal with the capability of the light load handling system to withstand earthquakes, the capability of shared equipment and components important to safety to perform their safety function in the event of a single failure, and the adequacy of the control system in limiting loads or load movement to prevent damaging the fuel to the extent that a release of radioactivity or a criticality accident could occur. The incorporation of the TPBARs does not necessitate a modification of the light load handling equipment. Since the external features of the TPBAR are the same as a burnable poison rod assembly (BPRA), the fuel assembly handling tools and the BPRA handling tool do not need to be modified. The impact of the TPBAR weight increase over the BPRA should be evaluated on a plant specific basis for this system.

SRP 9.1.5
Overhead Heavy Load Handling Systems: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4, 5 and 61. They deal with the capability of the overhead heavy load handling system to withstand earthquakes, the capability of shared equipment and components important to safety to perform their safety function in the event of a single failure, and the adequacy of the control system in limiting loads or crane load movement to prevent adverse effects on the function of essential equipment or a release of radioactivity. The overhead heavy load handling system is used in moving loads weighing more than one fuel assembly and its associated handling device (for the reference plant this is specified as <2000 lbs.). The incorporation of the TPBARs has no impact on this system. The spent fuel cask bridge crane is the only part of the overhead load handling system which might lift TPBAR assemblies. The scope of this report does not include the removal of the TPBARs from the spent fuel pool.

SRP 9.2.1
Station Service Water System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4, 5, 44, 45 and 46. They deal with the capability of the system to withstand the effects of earthquakes, missiles, pipe whip, jets, and environmental conditions resulting from high and moderate energy line breaks and dynamic effects associated with flow instabilities and loads; the capability of shared systems and components important to safety to perform required safety functions; the capability to transfer heat loads from safety-related structures, systems and components to an ultimate heat sink under normal and accident conditions; the provision for redundancy to assure that safety functions can be performed in the case of a single active component failure coincident with loss of offsite power; the capability to isolate components, subsystems or piping to preserve the safety function; the incorporation of design provisions to permit inservice inspection of safety-related components and equipment; and the incorporation of design provisions to permit operational functional testing of safety-related systems and components. The physical layout and operation of the Station Service Water

System are not required to be modified. However, the system heat transfer and flow requirements may be impacted by the TPC due to the increase in the spent fuel pool heat load during cooldown operations (see Section 2.9.3) and the subsequent impact on the Component Cooling Water System (see Section 2.9.4). Therefore, the impact on this system must be evaluated on a plant specific basis.

SRP 9.2.2
Reactor Auxiliary Cooling Water Systems: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4, 5, 44, 45 and 46. They deal with the capability of the system to withstand the effects of earthquakes, missiles, pipe whip, jets, and environmental conditions resulting from high and moderate energy line breaks and dynamic effects associated with flow instabilities and loads; the capability of shared systems and components important to safety to perform required safety functions; the capability to transfer heat loads from safety-related structures, systems and components to a heat sink under normal and accident conditions; the provision for redundancy to assure that safety functions can be performed in the case of a single active component failure coincident with loss of offsite power; the capability to isolate components, subsystems or piping to preserve the safety function; the capability to withstand loss of power without damage to RCP seals; design provisions to prevent fuel damage or reactor coolant leakage in excess of normal coolant-makeup capability in the event of a single failure; the incorporation of design provisions to permit inservice inspection of safety-related components and equipment; and the incorporation of design provisions to permit operational functional testing of safety-related systems and components. The physical layout and operation of the Cooling Water System are not required to be modified. However, the system heat transfer and flow requirements may be impacted by the TPC due to the increase in the spent fuel pool heat load during cooldown operations (see Section 2.9.3). Therefore, the impact on this system must be evaluated on a plant specific basis. A qualitative evaluation for the reference plant was performed and is discussed in Section 2.9.4.

SRP 9.2.3
Demineralized Water Makeup System: The acceptance criteria in this section are based on the relevant requirements of 10 CRF 50, Appendix A, GDC 2 and 5. They deal with the design of nonsafety-related systems such that failure due to a seismic event will not result in adverse effects on safety-related systems or components, and the design of shared systems such that sharing does not affect the safe shutdown of either unit. An evaluation of the potential impact of the TPC on this system is contained in Section 2.9.5.

SRP 9.2.4
Potable and Sanitary Water Systems: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 60. They deal with the design provisions provided to control the release of liquid effluents containing radioactive material from contaminating the Potable and Sanitary Water System (PSWS). The design of the PSWS is not being modified, therefore, the design features which prevented contamination of the PSWS in the reference plant (i.e., no cross-connection between the PSWS and any potentially radioactive system and the use of backflow prevention devices where plumbing

fixtures are located in areas susceptible to potential radiological hazard) are still present for the TPC plant. Therefore, there is no impact on this system.

SRP 9.2.5
Ultimate Heat Sink: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 5, 44, 45 and 46. They deal with the capability to withstand the effects of natural phenomena (i.e., earthquakes, tornadoes, hurricanes, and floods); the capability of shared systems and components to perform required safety functions; the capability to transfer heat loads from safety-related structures, systems, and components to the heat sink under normal and accident conditions; the provision for component redundancy to assure performance of safety functions assuming a single active component failure coincident with loss of offsite power; the capability to isolate components, systems, or piping so as not to compromise safety functions; the incorporation of design provisions to permit inservice inspection of safety-related components and equipment; and the incorporation of design provisions to permit operational functional testing of safety-related systems or components. The physical layout and operation of the Ultimate Heat Sink are not required to be modified. However, the heat removal requirement may be impacted by the TPC due to the increase in the spent fuel pool heat load during cooldown operations (see Section 2.9.3), and the subsequent impact on the Component Cooling Water System (see Section 2.9.4) and the Station Service Water System. Therefore, the impact on this system must be evaluated on a plant specific basis.

SRP 9.2.6
Condensate Storage Facilities: The acceptance criteria are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 5, 44, 45 and 46. They deal with the capability of the system to withstand the effects of natural phenomena (i.e., earthquakes, tornadoes, hurricanes, and floods); the capability of shared systems and components to perform required safety functions; the capability to provide sufficient makeup water to safety-related cooling systems; the provision for component redundancy to assure performance of safety functions assuming a single active component failure coincident with loss of offsite power; the capability to isolate components, systems, or piping so as not to compromise safety functions; the incorporation of design provisions to permit inservice inspection of safety-related components and equipment; and the incorporation of design provisions to permit operational functional testing of safety-related systems or components. The physical layout, design requirements and operation of the Condensate Storage Facility are not required to be modified for the TPC; therefore there is no impact on this system.

SRP 9.3.1
Compressed Air System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 1, 2 and 5. They deal with the design, fabrication and testing of systems and components important to safety to quality standards commensurate with the importance of the safety functions; the capability of the system to withstand the effects of earthquakes; and the capability of shared systems and components important to safety to perform required safety functions. For the reference plant, this system has no safety function. In addition, the TPC has no impact on the operation of the system.

SRP 9.3.2
Process and Post-Accident Sampling Systems: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 20.1(c) and 10 CFR 50, Appendix A, GDC 1, 2, 13, 14, 26, 41, 60, 63 and 64. They deal with the design of systems and components to standards commensurate with the importance of their safety functions; the capability of the system to withstand the effects of natural phenomena; the capability to monitor variables that can affect the fission process, the integrity of the core, and the reactor coolant pressure boundary; assuring the integrity of the reactor coolant pressure boundary; reliably controlling the rate of reactivity changes; reducing the concentration and quality of fission products released to the environment following postulated accidents; the capability of the system to control the release of radioactive materials to the environment; the ability to detect conditions that may result in excessive radiation levels in the fuel storage and radioactive waste systems; and the ability to monitor the containment atmosphere and plant environs for radioactivity. An evaluation of the sampling system was performed and is described in Section 2.9.6.

SRP 9.3.3
Equipment and Floor Drainage System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4 and 60. They deal with the capability of safety-related portions of the system to withstand the effects of earthquakes and flooding due to tank ruptures or pipe breaks, and the prevention of inadvertent transfer of contaminated fluids to a noncontaminated drainage system for disposal. The features of the existing system which allow it to meet these acceptance criteria, are not being modified for the TPC. The position of the TPBAR assembly within the fuel assembly will not cause contamination of noncontaminated fluids, therefore they do not impact this system.

SRP 9.3.4
Chemical and Volume Control System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 1, 2, 5, 14, 29, 33, 35, 60 and 61. They deal with the application of quality standards in accordance with the importance of the safety function; the capability to withstand earthquakes; the capability of shared systems and components to perform required safety functions; assuring reactor coolant pressure boundary material integrity by maintaining reactor coolant system water chemistry; the reliability of the CVCS to provide negative reactivity to the reactor by supplying borated water; the capability to supply reactor coolant makeup in the event of small breaks or leaks in the reactor coolant pressure boundary, to function as part of ECCS assuming a single active failure coincident with loss of offsite power; and provisions for confining radioactivity by venting and collecting drainage from the CVCS components through closed systems. Additional criteria defined in the SRP require provisions for monitoring the temperature upstream of the demineralizer and the filter demineralizer differential pressure, provisions for automatically diverting or isolating CVCS flow to the demineralizer, and establishing a leak reduction program. The physical design and operation of the CVCS are not being modified for the TPC. Because the RCS parameters are not affected by the TPC, the CVCS functional requirements for reactor coolant pump seal injection and RCS inventory control are not affected. Similarly, the safety injection function of the CVCS is not affected by the TPC, because the LOCA and other safety analyses are not

affected. Finally, the boration functions of the CVCS were evaluated against the core design boron requirements with the TPC, and were determined to be within the CVCS capabilities.

2.9.2 NEW AND SPENT FUEL STORAGE (SRP 9.1.1 AND 9.1.2) EVALUATION

2.9.2.1 Seismic Considerations

2.9.2.1.1 Introduction

Prior to loading into the reactor, the fuel assemblies with TPBAR assemblies will be placed in the new fuel storage area or in the spent fuel pool. After irradiation, the fuel and TPBAR assemblies will be stored in the spent fuel pool. The new and spent fuel storage racks where the assemblies will be stored are designed to meet plant specific seismic standards for the safe storage of the fuel. An evaluation has been performed to assess the impact of storing fuel assemblies with TPBAR assemblies in the reference plant new and spent fuel racks.

2.9.2.1.2 Methodology

The new and spent fuel racks are seismic Category I equipment and have previously been evaluated for seismic loadings for the case of storing standard fuel (i.e., fuel without TPBARs). The new fuel racks have been evaluated using plant specific response spectra and applying the response spectrum method of analysis. For the spent fuel racks, seismic loadings have been evaluated by performing nonlinear time history analysis.

An assessment of the impact of storing fuel with TPBARs in the reference plant new and spent fuel racks has been performed. The fuel assembly weights used in the previous seismic analyses have been reviewed, as well as considering whether the presence of the TPBAR assembly would have a significant effect on the fuel assembly's dynamic characteristics. The results of the assessment are presented in the following section.

2.9.2.1.3 Results

In the analyses previously performed for the reference plant storage racks, the weight used for a fuel assembly was a minimum of 1470 pounds. The combined weight of a reference plant fuel assembly with TPBARs inserted is less than 1431 pounds, which is a smaller weight than that previously evaluated. In addition, it has been determined that the TPBAR assembly which rests on the top nozzle adapter plate of the fuel assembly with the rodlets hanging in the thimble guide tubes has an insignificant effect on the fuel assembly's dynamic characteristics (see Section 2.4.2.1).

Since the new and spent fuel racks have previously been demonstrated to be acceptable for storage of assembly weights that are greater than that of a combined fuel assembly with TPBARs, and since the TPBAR assembly has an insignificant effect on the fuel assembly's dynamic characteristics, it is concluded that the existing seismic analysis of the fuel storage racks is valid for racks containing assemblies with TPBARs.

2.9.2.1.4 Potential Plant Specific Analytical Issues

The assessment performed was specific to the analyses previously performed for the reference plant new and spent fuel racks and the identified fuel assembly weights. Since fuel rack

evaluations are plant specific, the conclusion that the weights used in the analysis of record are greater than the weight of an assembly with TPBARs may not be applicable to all other plants. However, it is expected that for most (if not all) plants the weight of an assembly with TPBARs would not be significantly greater than that evaluated in the existing seismic analysis for the fuel storage racks.

2.9.2.2 Criticality Considerations

2.9.2.2.1 New Fuel Storage

The reference plant Technical Specifications (Reference 1) place a limit on the enrichment of fresh fuel that can be placed in the fresh fuel racks. The enrichment limit is 5.05 w/o ^{235}U. When enrichment uncertainties are taken into account, this becomes an effective limit of 5.00 w/o. The TPC designs described in Section 2.4.3 use fuel enrichments that are less than 5.00 w/o; the maximum enrichment of any fuel pin is 4.95 w/o. Thus, the TPC designs fall within the enrichment limit of the New Fuel Storage area for the reference plant. No credit is taken for the negative reactivity effect of burnable absorber components in this criticality analysis, so the TPBARs have no bearing on this enrichment limit.

2.9.2.2.2 Spent Fuel Storage

The reference plant has recently revised its spent fuel rack criticality analysis to include credit for soluble boron (Reference 2). In this analysis, various storage configurations are described, such as "all cell storage", "3-out-of-4 checkerboard storage", "2-out-of-4 checkerboard storage", and "3x3 checkerboard storage". These configurations define how assemblies of different reactivities can be placed in the spent fuel storage racks such that k_{eff} limits are met. In the Reference 2 analysis, enrichment credits are quantified for assembly burnup and IFBA (integral fuel burnable absorber), which allow fuel assemblies with higher enrichments to be placed in the various storage configurations if they meet the identified burnup and/or IFBA requirements. The 2-out-of-4 storage configuration allows fuel assemblies with up to 5.0 w/o enrichment to be stored in the spent fuel racks, with no credit for IFBA or burnup, as long as empty cells are face adjacent to the fuel assemblies. This configuration will accommodate the fuel enrichments of the TPC designs (maximum pin enrichment of 4.95 w/o). Depending on their burnup, some discharged assemblies from the TPC cores could be used in other storage configurations as well.

Because the number of feed assemblies is large for the TPC designs and the assembly discharge burnups are low relative to more typical fuel management, storage space within the spent fuel storage area would necessarily be used up more quickly if this kind of fuel management is implemented. Since fuel racks and available space differ from plant to plant, confirmation that a particular tritium production fuel management scheme will meet spent fuel storage k_{eff} limits will necessarily have to be done on a plant specific basis. In a plant specific analysis, storage configurations could be defined which reflect the fuel management employed and make best use of the fuel storage available space and the inventory of spent fuel assemblies.

2.9.2.2.3 References

1. Final Safety Analysis Report, Reference Plant.
2. WCAP-14720, Revision 3, (Non-Proprietary), "(Reference Plant) Spent Fuel Rack Criticality Analysis with Credit for Soluble Boron," Baker, M. M., et al., January 1998.

2.9.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM (SRP 9.1.3) EVALUATION

2.9.3.1 Introduction

The Spent Fuel Pool Cooling and Cleanup System provides the safety function of maintaining spent fuel pool cooling through the use of dual train pumps and heat exchangers, as well as the non-safety function of maintaining pool chemistry through the use of filters and demineralizers. The system consists of two cooling trains, each including a pump and heat exchanger. Normally, only one train is required to operate continuously to maintain the pool temperature within required limits. The heat exchangers are cooled by the safety-grade component cooling water system (see Section 2.9.4).

The original design basis heat load of a typical system included the insertion of 1/3 of a core for a 12-month cycle. Actual core offloads are plant-specific and vary in terms of the core fraction removed and the cycle length. However, a significant portion of the TPC core (~75%) will be removed during a normal refueling. This increases the heat load in the spent fuel pool and causes an increase in the maximum pool temperature.

2.9.3.2 Methodology

In general, the maximum spent fuel pool temperature is determined as follows:

1. Identify the number of fuel assemblies placed in the pool for each core cycle
2. Identify the length of each core cycle
3. Calculate the length of time each core fraction has been in the pool
4. Using an industry standard, determine the decay heat for each core fraction and add the decay heat values to calculate a total heat load
5. Split the heat load evenly between the two system heat exchangers
6. Using heat exchanger design data, back-calculate the minimum process inlet temperature (i.e., spent fuel pool temperature) that will transfer the heat load
7. Compare the calculated temperature to the maximum allowable temperature.

Westinghouse provided the Spent Fuel Pool Cooling System design for the reference plant, which is a typical 4-loop plant. Two key bases for sizing the system were: 1) removal of 1/3 of a core and 2) a 12-month fuel cycle. In addition, assumptions originally made about length of time to reload fuel and return to power (i.e., non-power time periods) are now known to be overly-conservative. The reference plant current licensing basis and operating experience is not consistent with these two assumptions. The reference plant has operated during refueling by removing about 100% of the core, performing various maintenance activities, then reinserting 60% of the core. In addition, decay heat standards have evolved since the original system performance calculations, in an attempt to more precisely model actual decay heat. In the case

of the reference plant, the heat load analysis is further complicated by moving spent fuel from one Unit's pool to the other.

In light of the above complications and because Westinghouse does not have the current heat load analysis of record, the analysis methodology will take two approaches: The first approach will examine two "best-estimate" cases. The first case is an estimate of the current heat load from Unit 1 in the Unit 1 pit, based on actual number of assemblies removed from the core for 18-month fuel cycles. The second case is based on removal of the required assemblies during a TPC equilibrium cycle. The plant uprating from 3411 to 3565 MWt core power will be included beginning with cycle 5. Heat loads from these two cases will be used in a heat exchanger performance calculation, assuming an average heat transfer coefficient, to determine a difference in maximum pool temperature between the two cases.

The heat load from each TPBAR is 3 watts at 150 hours after shutdown (Reference 3). Thus, even with as many as 3344 TPBARs in the core, the total additional heat load is trivial (0.003%) compared to the 3565 MWt core rating. Therefore, for the purposes of our calculations, the effect of the heat load from the TPBARs is within the accuracy of the overall calculation, and will be ignored. The heat load for each fuel assembly is assumed to represent an equal fraction of the total core heat load.

The second approach will utilize data from the reference plant FSAR, which documents the results of the current licensing basis for the system. Heat loads from this basis will be compared to the expected heat loads from the TPC.

2.9.3.3 Significant Input Parameters and Assumptions

Heat Exchanger Parameters

<u>Tube Side</u>	
Fluid	Spent fuel pool liquid
Flow rate	1,140,000 lb./hr. (design)
Inlet temp.	Calculated
Inlet press.	100 psia assumed
Resistance	6.37E-06 ft./gpm^2
<u>Shell Side</u>	
Fluid	Component cooling water
Flow rate	1,980,000 lb./hr. (design)
Inlet temp.	105°F (design; highest temp. is conservative)
Inlet press.	50 psia assumed
Resistance	1.98E-06 ft./gpm^2
<u>General</u>	
Heat transfer area	5470 ft^2
Heat transfer coefficient	Average of clean and service "u" factor $= (501 + 326)/2 = 413.5$ BTU/hr-ft^2 - °F

Heat Load

Decay heat loads for the first approach were derived from Reference 1. To simplify the analysis, decay heat was based on infinite burnup. This results in a higher than actual absolute heat load, but is adequate for purposes of determining the difference between the spent fuel pool temperature for the normal and TPC cores. The time increments were based on specific refueling data for the reference plant:

Δt_1 = average time from reactor shutdown to the end of defueling = 280.75 hr. = 1.01E06 sec.

Δt_2 = average time from end of defueling to 100% power = 673.0 hr. = 2.42E06 sec.

Cycle length = 1.5 yr. = 4.73E07 sec.

These assumptions result in the total cooling times shown in the following tables. The tables also show the fraction of the core removed at each shutdown.

Normal Core

Shutdown No.	Total Cooling Time (sec.)	Core Fract. Removed
1	3.04E+08	.435
2	2.54E+08	.435
3	2.03E+08	.373
4	1.52E+08	.435
5	1.01E+08	.420
6	5.07E+07	.435
7	1.01E+06	1.0

TPC Equilibrium Core

It is assumed that the 7th shutdown is now the TPBAR equilibrium cycle. In reality, there would be one or more transition cores. The analysis assumed an arbitrary number of cycles after the 7th shutdown. As shown later, the last shutdown (with the entire core removed) has the greatest effect on the total heat load.

Shutdown No.	Total Cooling Time (sec.)	Core Fract. Removed
1	5.07E+08	.435
2	4.57E+08	.435
3	4.06E+08	.373
4	3.55E+08	.435
5	3.04E+08	.420
6	2.54E+08	.435
7	2.03E+08	.7668
8	1.52E+08	.7668
9	1.01E+08	.7668
10	5.07E+07	.7668
11	1.01E+06	1.0

2.9.3.4 Acceptance Criteria

SRP 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," states that "For the maximum heat load with normal cooling systems in operation the temperature of the pool should be kept at or below 140°F..." The reference plant FSAR (Reference 2) licensing basis is rather complicated, and includes maximum temperatures with only single train cooling available. The original Westinghouse system design basis required that the pool be kept ≤ 120°F with two cooling trains and a normal 1/3 core offload, and ≤ 130°F for the case of a normal refueling followed by a full core offload immediately after attaining 100% power. The reference plant, which is typical of many plants, has modified the system design basis for heat loads, full core offloads (e.g., which is normal, not an emergency operation), etc., for consistency with current operating practices. Therefore, the SRP limit of 140°F will be used for the maximum heat load case.

2.9.3.5 Results

For the first approach, the cooling times listed in Section 2.9.3.3 were used, and the following decay heat loads were calculated using Reference 1.

Normal Core

Shutdown No.	Total Cooling Time (sec.)	Core Fract. Removed	Decay Heat (MWt)
1	3.04E+08	.435	0.344
2	2.54E+08	.435	0.382
3	2.03E+08	.373	0.386
4	1.52E+08	.435	0.512
5	1.01E+08	.420	0.569
6	5.07E+07	.435	2.207
7	1.01E+06	1.0	10.690
TOTAL			15.091

TPC Equilibrium Core

It is assumed that the 7th shutdown is now the TPC equilibrium cycle. In reality there would be one or more transition cores.

Shutdown No.	Total Cooling Time (sec.)	Core Fract. Removed	Decay Heat (MWt)
1	5.07E+08	.435	0.150
2	4.57E+08	.435	0.253
3	4.06E+08	.373	0.228
4	3.55E+08	.435	0.309
5	3.04E+08	.420	0.347
6	2.54E+08	.435	0.400
7	2.03E+08	.7668	0.830
8	1.52E+08	.7668	0.944
9	1.01E+08	.7668	1.031
10	5.07E+07	.7668	3.891
11	1.01E+06	1.0	10.690
TOTAL			19.073

The maximum pool temperature with the normal core design at the end of Cycle 7 was calculated to be 138°F, and with the TPC core, 147°F, at the end of Cycle 11. Thus, it is expected that the 140°F limit would be exceeded.

It can be seen that there are many parameters which can easily be varied, justifiably, to affect the results of the evaluation. For example, no uncertainty was applied to the ANS standard decay heat loads (this is non-conservative). More limiting time periods could be applied to the refueling schedule. For example, a minimum of 100 hours (3.6E05 sec.) after reactor shutdown is required by Technical Specifications before fuel can be moved. The analysis included an average of about 142 hours from offline to refueling mode, based on actual experience. Although a small impact, this assumption is also non-conservative.

Using the FSAR data for the second approach results in the data shown in Table 2.9.3-1. In all these cases, only one spent fuel pool heat exchanger is available for heat removal. The table shows that the effect of the fuel management scheme of the TPC is an increase in the maximum spent fuel pool temperature by up to 21°F, depending upon the case.

2.9.3.6 Conclusions and Affected Documents

Although the bases for the calculated absolute temperatures may be debated, the increase in temperature (up to 21°F) is not trivial. There is definitely a negative impact on system heat removal capability. It is recommended that the spent fuel pool cooling system analysis of record for any plant considering incorporation of TPBARs, be reviewed in detail to determine if licensing basis limits will be exceeded. As a minimum, the FSAR description of the spent fuel pool cooling system will need to be revised to reflect the new inputs and the resulting maximum pool temperatures. In the worst case, changes may be necessary to the system capacity in order to cope with higher heat loads.

2.9.3.7 References

1. ANSI/ANS 5.1-1994, "Decay Heat Power in Light Water Reactors".

2. Reference Plant, Final Safety Analysis Report Update

3. PNNL-11419, Rev. 1, "Report on the Evaluation of the Tritium Producing Burnable Absorber Rod Lead Test Assembly", (Section 6.2.1.2), March 1997, Pacific Northwest Laboratory, Richland, Washington.

Table 2.9.3-1 Effect of TPBAR on Spent Fuel Pool Cooling System Assessment Based on Reference Plant FSAR**								
Case	**Previous Assemblies**	**Previous Assemblies Heat Load (MBTU/hr.)**	**New Offload**	**New Offload Heat Load (MBTU/hr.)**	**SFP Pump Heat Load (MBTU/hr.)**	**Margin (MBTU/hr.)**	**Total Heat Load (MBTU/hr.)**	**Max. SFP Temp. (°F)**
Normal	2014 (1006 - U2 + 1008 - U1)	11.96	84 assem. @ 150 hr.	16.65	0.38	0.50	29.49	139.3
Normal with TPBAR	2014 (1006 - U2 + 1008 - U1)	11.96	148 assem. @ 150 hr.	(148/84)(16.65) = 29.33	0.38	0.50	42.18	160
Max. Normal	2014 (1006 - U2 + 1008 - U1)	≤ 11.98	1 core @ 120 hr.	41.24*	0.38	0.50	54.1	171.1
Max. Normal with TPBAR	>2014 (1006 - U2 + 1008 - U1)	>11.98	1 core @ 120 hr.	41.24*	0.38	0.50	>54.1	>171.1
Max. Emerg.	1821 (897 - U2 + 924 - U1)	≤ 11.18	84 assem. @ 36 days + 1 core @ 150 hr.	~ 10 + 36.07 = 46.07*	0.38	0.50	58.13	182
Max. Emerg. With TPBAR	1821 (897 - U2 + 924 - U1)	≤ 11.18	148 assem. @ 36 days + 1 core @ 150 hr.	~ [(148/84)(10) =17.62] + 36.07 = 53.62	0.38	0.50	65.7	190

* New offload heat load = total heat load (from FSAR**) - spent fuel pool pump heat (from FSAR**) - margin (from FSAR**) - previous assemblies heat load (from FSAR**)

** (Reference 2)

2.9.4 COMPONENT COOLING WATER SYSTEM (SRP 9.2.2) EVALUATION

The component cooling water system for the reference plant is a unique design (with respect to the cooling loop separation) compared to similar systems provided by Westinghouse. The component cooling water system is designed to provide cooling water only to the residual heat removal heat exchangers, the motor coolers for the residual heat removal pumps, and the spent fuel pool heat exchangers. The component cooling water heat exchangers are cooled by a separate nuclear service cooling water system.

During plant refueling, more fuel assemblies are replaced in the core when TPBARs are incorporated (see Section 2.9.3). This would appear to increase the heat load on the component cooling water system. However, this increase in heat load is offset by a corresponding reduction in heat load from the residual heat removal system heat exchangers, because there is less decay heat coming from the RCS, since there are fewer assemblies in the reactor. Thus, there is no net effect on the component cooling water system during refueling.

However, for plant cooldown, there are two analyses of record. The first is normal plant cooldown from 350°F to 140°F. This case assumes both residual heat removal and component cooling water heat exchangers are available. The design basis cooldown time limit is 20 hours after reactor shutdown. The second analysis is referred to as single train cooldown from 350 to 200°F, in which only one residual heat removal and one component cooling water heat exchanger are available. For this case, the cooldown time limit is 36 hours.

In both cases, the heat load from the spent fuel pool heat exchangers is assumed to be approximately 29.5 MBTU/hr. This heat load includes 16.65 MBTU/hr from the most recent refueling (see Table 2.9.3-1), based on decay heat at 150 hr after shutdown. This heat load is very conservative, because the 150 hour time period does not take credit for a significant length of time (days) required to refuel the reactor, install the reactor vessel head, refill, pressurize, heatup, and perform other startup activities before the plant could then be shut down and cooled with the residual heat removal and component cooling water systems. This additional time will significantly reduce the actual spent fuel pool heat load at the time of cooldown. Nevertheless, the increased heat load from the spent fuel pool heat exchangers from 29.5 to 42 MBTU/hr will have an adverse impact on the ability of the component cooling water and nuclear service cooling water systems to cool the plant within the design basis time period. The extent of this impact will depend upon available margins in the systems designs, if any, and should therefore be evaluated on a plant-specific basis.

During normal plant power operation, the increased heat load from the spent fuel pool heat exchangers will tend to increase the normal component cooling water temperature. However, this case is normally less limiting for the component cooling water system than the cooldown cases described above.

2.9.5 DEMINERALIZED WATER MAKEUP SYSTEM (SRP 9.2.3) EVALUATION

Due to the incorporation of TPBARs, the tritium levels in the reactor coolant system will increase, as discussed in Section 2.11. To maintain the tritium concentration within acceptable levels, the normal discharge volume of processed reactor coolant to the environment over the

course of a core cycle will increase. Therefore, the makeup water requirements will also increase. In most plants, the demineralized water system provides makeup to a primary water storage tank which then feeds the chemical and volume control system makeup control system. Thus, the need for makeup from the demineralized water system will increase. It is expected that the system will be operated more frequently, and that equipment sizing and flow rates will not be affected. However, the impact of increased demand must be evaluated on a plant-specific basis.

2.9.6 PROCESS SAMPLING SYSTEM (SRP 9.3.2) EVALUATION

2.9.6.1 Introduction

The purpose of the Process Sampling System is to collect and route liquid and gaseous sample fluid from various NSSS systems to either a local sampling station or the sample room for collection and analysis. Sample fluid is cooled and depressurized as required for safe handling by the operator.

2.9.6.2 Methodology

Based upon a review of TPBAR sampling requirements beyond the normal plant sampling requirements, a review of the sampling system design was conducted to determine if additional sample connections would be required from new or existing systems which do not currently include a sample tap, or if existing local sample lines must be routed to the sample room.

2.9.6.3 Results and Conclusions

There will be an increased focus on tritium and lithium analyses, but from the existing RCS liquid sampling lines to the sample room.

Due to the introduction of TPBARs to the spent fuel pit, the potential exists for leaching of chemical contaminants, such as aluminum. Spent fuel pool contents are sampled locally periodically to verify that the minimum boron concentration is maintained, and that the maximum levels of contaminants are not exceeded. The existing sampling frequency should be increased during, and immediately after, refueling with TPBARs to assure compliance with existing spent fuel pool specifications. It is possible that analyses for all chemicals, including tritium, could be performed with the same sample.

No additional sample points have been identified, so there is no impact on the sampling system design.

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2.10 STEAM AND POWER CONVERSION SYSTEM

The sections in Chapter 10 of the SRP deal with the secondary side systems for steam and power conversion. Based on the following qualitative discussions, it was determined that further evaluations of potential TPBAR impact were not required, and it is not anticipated that there would be changes to Chapter 10 of a typical SAR as a result incorporating TPBARs.

SRP 10.2
Turbine Generator: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 4. They deal with the protection of structures, systems, and components important to safety from the effects of turbine missiles by providing a turbine overspeed protection system with suitable redundancy. The TPBARs have no interaction with the turbine overspeed protection system, therefore incorporation of the TPBARs will have no impact on the operation of the system.

SRP 10.2.3
Turbine Disk Integrity: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 4. They deal with the protection of structures, systems, and components important to safety from the effects of missiles that may result from equipment failure. These criteria are met through the use of materials with acceptable fracture toughness and elevated temperature properties, adequate design, and requirements for preservice and inservice inspections. The TPBARs have no impact on the NSSS performance parameters and, therefore, will have no impact on the turbine.

SRP 10.3
Main Steam Supply System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4, 5 and 34. They deal with the capability of the safety-related portions of the system to withstand the effects of natural phenomena, external missiles, internally generated missiles, pipe whip, and jet impingement forces; precautions to avoid steam hammer, relief valve discharge loads, and water entrainment; the capability of shared systems and components to perform required safety functions; and the capability to transfer residual and sensible heat from the reactor system. The TPBARs have no impact on the NSSS performance parameters and, therefore, will have no impact on the main steam supply system.

SRP 10.3.6
Steam and Feedwater System Materials: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.55a; 10 CFR 50, Appendix A, GDC 1 and 35; and 10 CFR 50, Appendix B. They deal with the design, fabrication, construction, testing and inspection of structures, systems and component commensurate with the importance of the safety function to be performed; the provision of a system to supply emergency core cooling such that reactor core component damage is minimal following any loss of reactor coolant; the structural integrity of pressure containing components including freedom from brittle fracture; and the appropriate level quality assurance. The TPBARs have no impact on the NSSS performance parameters and, therefore, will have no impact on the steam and feedwater system materials.

SRP 10.4.1

Main Condensers: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 60. They deal with the design of the system such that failures do not cause excessive releases of radioactivity to the environment, unacceptable condensate quality or flooding of areas housing safety-related equipment. The TPBARs have no impact on the condensate quality or potential for flooding. The TPBARs do not have direct contact with the secondary water. Operation with TPBARs does result in an increase in the tritium released to the primary coolant but this increase is countered by an increased discharge of primary coolant to prevent the tritium level in the coolant from exceeding the design basis level. Thus, the radiological aspects of primary-to-secondary leakage are not impacted by the TPBARs. Consequently, there is no impact on the Main Condensers.

SRP 10.4.2

Main Condenser Evacuation System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 60 and 64. They deal with the design of the system such that failures do not cause excessive releases of radioactive materials to the environment, and the provision for monitoring releases of radioactive materials. The design of this system is not changing and the radioactivity monitor currently existing for the discharge of the condenser mechanical vacuum pumps will continue to be used. The TPBARs do not have direct contact with the secondary water. Operation with TPBARs does result in an increase in the tritium released to the primary coolant but this increase is countered by an increased discharge of primary coolant to prevent the tritium level in the coolant from exceeding the design basis level. Thus, the radiological aspects of primary-to-secondary leakage are not impacted by the TPBARs. Consequently, there is no impact on the Main Condenser Evaluation System.

SRP 10.4.3

Turbine Gland Sealing System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 60 and 64. They deal with the design of the system such that failures do not cause excessive releases of radioactive materials to the environment, and the provision for monitoring releases of radioactive materials. The design of this system is not being changed and the radioactivity monitor currently existing in the discharge line will continue to be used. The TPBARs do not have direct contact with the secondary water. Operation with TPBARs does result in an increase in the tritium released to the primary coolant but this increase is countered by an increased discharge of primary coolant to prevent the tritium level in the coolant from exceeding the design basis level. Thus, the radiological aspects of primary-to-secondary leakage are not impacted by the TPBARs. Consequently, there is no impact on the Turbine Gland Sealing System.

SRP 10.4.4

Turbine Bypass System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 4 and 34. They deal with the design of the system such that a failure due to a pipe break or malfunction of the system will not adversely affect essential systems or components; and the ability to use the system for shutting down the plant during normal operations. The design of this system is not being

changed and the TPBARs have no impact on the NSSS performance parameters. Therefore, there is no impact on the Turbine Bypass (steam dump) System.

SRP 10.4.5
Circulating Water System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 4. They deal with the prevention or detection and control of flooding in safety-related areas due to leakage from the system; and the design such that malfunction or failure of a component or piping of the system (including an expansion joint) will not have unacceptable effects on the performance of safety-related systems or components. The design of this system is not being changed and the TPBARs have no interface with the circulating water system. Therefore, there is no impact on the Circulating Water System.

SRP 10.4.6
Condensate Cleanup System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 14. They deal with the control of secondary water chemistry in order to prevent degradation of the primary coolant boundary integrity. The TPBARs do not have direct contact with the secondary water and thus would not impact the chemistry. Operation with TPBARs does result in an increase in the tritium released to the primary coolant but this increase is countered by an increased discharge of primary coolant to prevent the tritium level in the coolant from exceeding the design basis level. Thus, the radiological aspects of primary-to-secondary leakage are not impacted by the TPBARs. Consequently, there is no impact on the Condensate Cleanup System.

SRP 10.4.7
Condensate and Feedwater System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4, 5, 44, 45 and 46. They deal with the capability of the safety-related portions of the system to withstand the effects of earthquakes; precautions to avoid water hammer; the capability of shared systems and components to perform required safety functions; the capability to transfer heat loads from the reactor system to a heat sink under normal and accident conditions; the redundancy of components to assure that the safety function can be performed under accident conditions assuming a single active component failure; the capability to isolate components, subsystems or piping if required; the provision to permit periodic inservice inspection; and the provision to permit functional testing. The design of this system is not being changed and the TPBARs have no impact on the NSSS performance parameters. Therefore, there is no impact on the Condensate and Feedwater System.

SRP 10.4.8
Steam Generator Blowdown System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 1, 2 and 14. They deal with the use of appropriate quality standards for system components; the capability of the system to withstand the effects of earthquakes; and control of secondary water chemistry in order to prevent degradation of the primary coolant boundary integrity. The TPBARs do not have direct contact with the secondary water and thus would not impact the chemistry. Operation with TPBARs does result in an increase in the tritium released to the primary

coolant but this increase is countered by an increased discharge of primary coolant to prevent the tritium level in the coolant from exceeding the design basis level. Thus, the radiological aspects of primary-to-secondary leakage are not impacted by the TPBARs. Consequently, there is no impact on the Condensate Cleanup System.

SRP 10.4.9
Auxiliary Feedwater System: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 2, 4, 5, 19, 34, 44, 45 and 46. They deal with the capability of the safety-related portions of the system and structures housing the system to withstand the effects of earthquakes, external missiles, internally generated missiles, pipe whip and jet impingement forces; the capability of shared systems and components to perform required safety functions; the capability of system instrumentation and controls for prompt hot shutdown and potential capability for subsequent cold shutdown; the capability to transfer heat loads from the reactor system to a heat sink under normal and accident conditions; the redundancy of components to assure that the safety function can be performed under accident conditions assuming a single active component failure; the capability to isolate components, subsystems or piping if required; the provision to permit periodic inservice inspection; and the provision to permit functional testing. The design of this system is not being changed and the TPBARs have no impact on the NSSS performance parameters. Therefore, there is no impact on the auxiliary feedwater system.

2.11 RADIOACTIVE WASTE MANAGEMENT

2.11.1 INTRODUCTION

The sections of Chapter 11 of the SRP address the management of radioactive waste. The impact of the TPBARs on the source term is described in Section 2.11.2. The impacts on the liquid, gaseous, and solid waste management systems are described in Sections 2.11.3 through 2.11.5, and Section 2.11.6 discusses the impact on the radiological monitoring and sampling systems.

SRP 11.1
Source Terms: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 20; 10 CFR 50, Appendix A, GDC 60; and 10 CFR 50, Appendix I. They deal with the radioactivity in effluents to unrestricted areas; design objectives and limiting conditions for operation to meet the ALARA criterion; and the control of radioactive releases to the environment. An evaluation of the impact of the TPBARs on the source terms is discussed in Section 2.11.2.

SRP 11.2
Liquid Waste Management Systems: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 20.106; 10 CFR 50.34a; 10 CFR 50, Appendix A, GDC 60 and 61; and 10 CFR 50, Appendix I, Sections II.A and II.D. They deal with the radioactivity in effluents to unrestricted areas; design objectives and limiting conditions for operation to meet the ALARA criterion; the presentation of sufficient design information to demonstrate that design objectives to control releases of radioactive effluents have been met; the control of radioactive releases to the environment; and the design of the system to assure adequate safety under normal and postulated accident conditions. An evaluation of the impact of the TPBARs on the liquid waste management is discussed in Section 2.11.3.

SRP 11.3
Gaseous Waste Management Systems: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 20.106; 10 CFR 50.34a; 10 CFR 50, Appendix A, GDC 3, 60 and 61; and 10 CFR 50, Appendix I, Sections II.B, II.C and II.D. They deal with the radioactivity in effluents to unrestricted areas; design objectives and limiting conditions for operation to meet the ALARA criterion; the presentation of sufficient design information to demonstrate that design objectives to control releases of radioactive effluents have been met; the control of radioactive releases to the environment; protection from the effects of an explosive mixture of hydrogen and oxygen; and the control of radioactivity in gaseous waste management systems and ventilation systems associated with fuel storage and handling areas. An evaluation of the impact of the TPBARs on the gaseous waste management is discussed in Section 2.11.4.

SRP 11.4
Solid Waste Management Systems: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 20.106; 10 CFR 50.34a; 10 CFR 50, Appendix A, GDC 60, 63 and 64; and 10 CFR 71. They deal with the radioactivity in effluents to unrestricted areas; the presentation of sufficient design information to demonstrate that design objectives to

control releases of radioactive effluents have been met; the control of radioactive releases to the environment; the provision for monitoring radiation levels and leakage; and radioactive material packaging. An evaluation of the impact of the TPBARs on the solid waste management is discussed in Section 2.11.5.

SRP 11.5
Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 20.106 and 10 CFR 50, Appendix A, GDC 60, 63 and 64. They deal with radioactivity monitoring of effluents to unrestricted areas; the control of radioactive releases to the environment; and the provision for monitoring radiation levels and leakage. An evaluation of the impact of the TPBARs on these systems is discussed in Section 2.11.6.

2.11.2 SOURCE TERMS (SRP 11.1)

2.11.2.1 Acceptance Criteria

The acceptance criteria for the radioactive source terms includes a review of design basis source terms to determine that the expected releases associated with applicable regulations are met. These regulations are 10 CFR Part 20 as it relates to radioactivity in effluents to unrestricted areas, 10 CFR Part 50 - Appendix I as it relates to "as low as is reasonably achievable" criterion, and GDC 60 as it relates to waste management systems and the control of releases of radioactive materials to the environment.

2.11.2.2 Evaluation

The following sources have been reviewed to assess the impact of TPBARs on the source terms:

- Design Basis Sources
- Realistic Sources
- Tritium Sources

Design Basis Sources

Gamma ray sources considered in the plant design include fission and corrosion product sources, as well as activation sources such as the nitrogen-16 activity in the primary coolant. The corrosion product and nitrogen-16 sources are not expected to be significantly affected by the core design differences between a TPC and a conventional core. However, the somewhat higher enrichments and reduced fuel assembly burnups associated with the TPC core design, as compared to conventional core designs, can influence the design basis source terms. This has a potential impact on the assessment of waste management systems for treatment of liquid and gaseous wastes.

In order to quantify the impact of the core design differences, the ORIGEN2 computer code (Reference 1) was used to evaluate the TPC fission product nuclide inventories in the reactor core for the reference plant. The major fission product activity concentrations in the primary

coolant were also evaluated. These activities were compared to similar calculations made for the reference plant operating with a conventional core. The results of the evaluation are summarized below.

Table 2.11.2-1 lists the major core noble gas and iodine isotopic inventories at reactor shutdown for the reference plant operating with a conventional core and with a TPC. Calculations for the TPC were performed for the initial and equilibrium cycles and the maximum isotopic inventory of both cycles was selected to create a composite set of core isotopic inventories for a TPC. It is noted that the total inventory of the iodine activities is the same or lower in the TPC core as compared to a conventional core. The noble gas inventories tend to exhibit a less consistent pattern, with some isotope inventories being the same or lower in the TPC and some being higher. However, the increases are generally small and in the range of 10-15%. Differences of this magnitude may be ascribed to the conservatism introduced in the TPC calculations by considering the maximum values from both the initial and equilibrium cycles and/or ORIGEN modeling differences. Further, the activity inventories for the more important noble gases are less than those associated with a conventional core; i.e. the long-lived Kr-85 is about 30% lower in the TPC as compared to the conventional core and the most abundant Xe-133 is 5% lower in the TPC.

Table 2.11.2-2 includes the calculated RCS coolant activities for noble gases and iodines for the equilibrium cycle of a conventional core and the TPC. Again, for conservatism, the TPC activities were maximized in that the greater concentration associated with the initial and equilibrium cycles was considered to create a composite set of RCS activities. The results of a comparison of the values are similar to those for the core activities; that is, total inventory of the iodine activities is the same or lower in the TPC core as compared to a conventional core and the noble gas inventories exhibit a less consistent pattern, with some isotopes inventories being the same or lower in the TPC and some being higher, but differences generally limited to less than 15%.

Based on the above comparisons, the impact of operation with a TPC is expected to have a negligible impact on the design basis sources and the treatment of fission products in liquid and gaseous waste. The small differences noted in the fission product activities associated with a TPC operation are not expected to affect the ability of the plant to meet the applicable regulatory requirements relative to radioactivity in effluents to unrestricted areas (10 CFR Part 20), the "as low as is reasonably achievable" criterion (10 CFR Part 50 - Appendix I), and waste management systems and the control of releases of radioactive materials to the environment (GDC 60).

Realistic Source Terms

Realistic plant activity concentrations in the RCS, secondary side liquid, and secondary side steam as reported in plant Safety Analysis Reports (SARs) are based on ANSI/ANS 18.1-1984 (Reference 2) or its predecessor ANSI-N237-1976 (Reference 3). Corrections to the default data reported in the standards are made if warranted by differences in plant-specific parameters. For example, the realistic source terms for the reference plant are based on the application of correction factors to the data in ANSI/ANS 18.1-1984, including the assumed operation of a

Westinghouse gaseous waste management system. These adjustment factors for the realistic fission and corrosion products do not change based on operation with TPCs.

There are no adjustment factors in the ANS standards for converting the default tritium concentrations to plant-specific values. As noted in the standards:

> "The concentration of tritium is a function of the inventory of tritiated liquids in the plant, the rate of production of tritium due to activation in the reactor coolant, the rate of release from the fuel, and the extent to which tritiated water is recycled or discharged from the plant. The tritium concentration given ... is representative of PWRs with a moderate amount of tritium recycle."

Although operation with a TPC could increase the amount of tritium produced over a particular fuel cycle, the recommended waste management solution to accommodating an increase is to discharge the additional inventory in plant effluents in order to limit concentration buildup in plant water volumes and minimize occupational radiation exposures within the plant. This is the current practice at operating PWR plants and evaluations indicate that in-plant tritium concentrations can be maintained at current operating plant levels without compromising release limits and dose to members of the public. Thus, TPC operation does not impact the realistic source term activity concentrations. The impacts to the plant staff and to reactor operations of retaining tritium within the facility are discussed in Section 2.11.3.2. It should also be noted that, with the exception of some tritium decay, all of the tritium will be eventually released to the environment.

If the tritium inventory released to the primary coolant increases and in-plant concentrations are maintained at current levels, the expected tritium activity discharges can be expected to increase. The activity concentrations that are developed from the methodology associated with the ANS standards are also used in the GALE computer code (Reference 4) which is used in estimating the activity of fission and corrosion products in gaseous and liquid effluents. Although the parameters and assumptions that are input to the code may not be affected by operation with a TPC, an assumption inherent to the GALE code is that the tritium releases through the combined liquid and vapor pathways are 0.4 Ci/yr per MWt. This release rate is based on a review of the tritium release rates at a number of operating PWRs. However, this data reflects operation with conventional core designs and since the TPC has the potential for a higher release rate, the tritium release in plant effluents should be updated. The projected increases in plant tritium activity production and effluents are described in the following section.

Tritium Source Terms

<u>Sources for Conventional Cores</u>

Tritium is currently generated in commercial nuclear plants and is a component of waste effluents from the operating plants. The tritium is produced as a ternary fission product in the fuel as well as by neutron interactions with boron in core components and boron, lithium, and deuterium in the primary coolant. Specifically, the sources of tritium in operating PWRs are,

- fission-produced tritium
- production from soluble boron reactions
- production from soluble lithium reactions
- production from burnable poison rods and Integral Fuel Burnable Absorbers (IFBAs)
- production from control rods
- production from deuterium reactions

Most of the tritium in the plants is produced as a ternary fission product whose source is fission in the fuel. Tritium from this source, as well as the tritium produced from boron reactions in burnable poison rods, IFBAs, and control rods, must diffuse through cladding before release to the RCS. Tritium from the other sources identified above are produced directly in the RCS through nuclear reactions involving boron, lithium, and deuterium.

Design Basis Tritium Sources from TPBAR Permeation

A design objective of the TPBARs is to retain as much tritium as possible within the TPBAR. However, a small quantity of tritium may permeate the cladding material into the primary coolant. Even though the amount could be only a small fraction of the total amount that is generated within the TPBAR, the possibility of a significant release cannot be discounted and the additional contribution to the plant tritium inventory must be considered. For design purposes, a goal of less than 1.0 curie per year per TPBAR release to the primary coolant has been established. The expected value is less than 1 curie per year and will be determined from appropriate monitoring and surveillance programs described in Section 2.11.3.2, "Evaluation of TPBAR Failures". In addition, the equilibrium cycle TPC design includes more than 14,000 IFBA rods that must be considered as an additional source of tritium in the primary coolant.

The potential contribution of the TPBARs on the plant tritium source term was evaluated and compared to that without TPBARs. A review of current operating plant data was also performed as part of this evaluation in order to verify the current design basis tritium source terms. Results are listed in Table 2.11.2-3.

As noted in the table, the total amount released to the reactor coolant system of a representative 4-loop Pressurized Water Reactor (PWR) is approximately 890 Ci per year. This value is based on the average of reported tritium releases (Reference 5) from mature operating plants for the period 1974 to 1993. Thus, it is assumed that the amount released is roughly equivalent to the amount produced in plants that have operated beyond a few fuel cycles. As indicated in Figure 2.11.2-1 the equilibrium release rate (equated to the production rate) of tritium is approximately 0.25 Ci/yr-MWt. For a 4-loop plant operating at 3565 MWt, the nominal production rate is 890 Ci/yr. The value is consistent with the peak range of 800-900 Ci per year observed at the reference plant and is considered to be the value associated with a conventional core in evaluations of the impact of TPC on plant radiological conditions and waste management.

Operation of a commercial nuclear plant with TPBARs has the potential for increasing the tritium activity in the primary coolant system. Based on a design goal release of 1 Ci/rod per year from TPBARs for a plant operating a full complement of TPBARs, the increase would be 3,344 curies of tritium per year from TPBARs and another 34 curies from IFBA fuel rods. Thus, the total annual production could potentially increase from a nominal value of 890 Ci/year to 4,268 Ci/year or roughly a five-fold increase in the primary coolant tritium inventory.

<u>Tritium Sources for TPC From Postulated TPBAR Failures</u>

In addition to the design basis release from the TPBARs by permeation processes, an additional potential release scenario to be considered is the failure of one or more of the TPBARs, such that the inventory in the TPBAR is released to the primary coolant. Note that the tritium is not in a gaseous form and would have to be leached from the pellet and getter. The assumed failure rate for this release scenario is the simultaneous failure of two rods. The maximum amount of tritium generated in any single TPBAR is 1.05 grams at the end of the initial cycle and 1.044 at the end of the equilibrium cycle, which is equivalent to 10,000 Ci per rod or a total of 20,000 Ci.

The failure of TPBARs with postulated subsequent release of all tritium is considered to be beyond that associated with reasonable design basis considerations. Thus, detailed evaluations have focused on the "permeation" source as the design basis tritium source term for operation with a TPC. However, the impact of two failed rods will be addressed (see Section 2.11.3.2) to ascertain the general impact of this abnormal event on the ability to remain within applicable regulatory limits. It is found that the reference plant is capable of safe operation with a TPC and that plant releases can be maintained within prescribed regulatory limits. The projected impact on liquid, gaseous, and solid waste is found to be small in terms of off-site dose consequences and generation of solid waste.

2.11.2.3 Conclusions

Based on the above comparisons, the impact of operation with a TPC is expected to have a negligible impact on the design basis and realistic fission and corrosion product sources and the treatment of these isotopes in liquid and gaseous waste. The design basis tritium sources are expected to increase the amount of tritium that is discharged annually by a factor of about five.

As demonstrated in Section 2.11.3.2, the differences noted in the source terms for TPC operation would not affect the ability of the plant to remain well within the applicable regulatory requirements relative to radioactivity in effluents to unrestricted areas (10 CFR Part 20), the "as low as is reasonably achievable" criterion (10 CFR Part 50 - Appendix I), and waste management systems and the control of releases of radioactive materials to the environment (GDC 60).

2.11.2.4 References

1. ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Material, Allen G. Croff, Nuclear Technology, Vol. 62, September 1983.

2. ANSI/ANS-18.1-1984, American National Standard - Radioactive Source Term for Normal Operation of Light Water Reactors, approved December 31, 1984.

3. American National Standard N237 - 1976 (ANS-18.1), Source Term Specification, approved May 11, 1976.

4. NUREG-0017, Rev. 1, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors, Division of System Integration, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, April 1985.

5. NUREG/CR-2907, Radioactive Materials Released from Nuclear Power Plants.

Table 2.11.2-1
Comparison of Core Noble Gas and Iodine Activities for a Conventional Core to a Tritium Producing Core

Isotope	Total Core Activity at Shutdown (Curies)	
	Conventional Core	TPC
I-131	9.9×10^7	9.3×10^7
I-132	1.4×10^8	1.4×10^8
I-133	2.0×10^8	1.9×10^8
I-134	2.2×10^8	2.1×10^8
I-135	1.9×10^8	1.8×10^8
Kr-85	8.9×10^5	6.9×10^5
Kr-85m	2.6×10^7	2.8×10^7
Kr-87	4.7×10^7	5.4×10^7
Kr-88	6.8×10^7	7.6×10^7
Xe-133	2.0×10^8	1.9×10^8
Xe-135	4.5×10^7	4.7×10^7
Xe-135m	4.0×10^7	3.7×10^7
Xe-138	1.6×10^8	1.6×10^8

Table 2.11.2-2
Comparison of Reactor Coolant Noble Gas and Iodine Activities for a Conventional Core to a Tritium Producing Core

Isotope	RCS Activity Concentration (μCi/g)	
	Conventional Core	TPC
I-129	5.4 x 10^{-8}	4.1 x 10^{-8}
I-130	2.7 x 10^{-2}	2.6 x 10^{-2}
I-131	2.8	2.7
I-132	2.8	2.7
I-133	5.3	4.1
I-134	6.6 x 10^{1}	5.6 x 10^{-1}
I-135	2.6	2.2
Kr-83m	4.7 x 10^{-1}	4.6 x 10^{-1}
Kr-85m	2.0	1.9
Kr-85	7.1	7.4
Kr-87	1.2	1.2
Kr-88	3.5	3.6
Kr-89	1.0 x 10^{-1}	9.7 x 10^{-2}
Xe-133	2.4 x 10^{2}	2.7 x 10^{2}
Xe-135	8.0	8.3
Xe-135m	5.4 x 10^{-1}	4.5 x 10^{-1}
Xe-137	2.0 x 10^{-1}	1.6 x 10^{-1}
Xe-138	7.1 x 10^{-1}	6.1 x 10^{-1}

Table 2.11.2-3
Annual Tritium Produced in the Primary Coolant (Curies)

Maximum expected annual releases to RCS (typical 4-loop PWR)*	890
Additional annual inventory from TPC	
- TPBARs	3344
- IFBAs	34
Typical 4-loop with conventional core	**890**
Typical 4-loop with TPC	**4268**

* - Assumes 0.25 Ci/MWt-year, which is based on data from operating plant Annual Radioactive Release reports and a power level of 3565 MWt

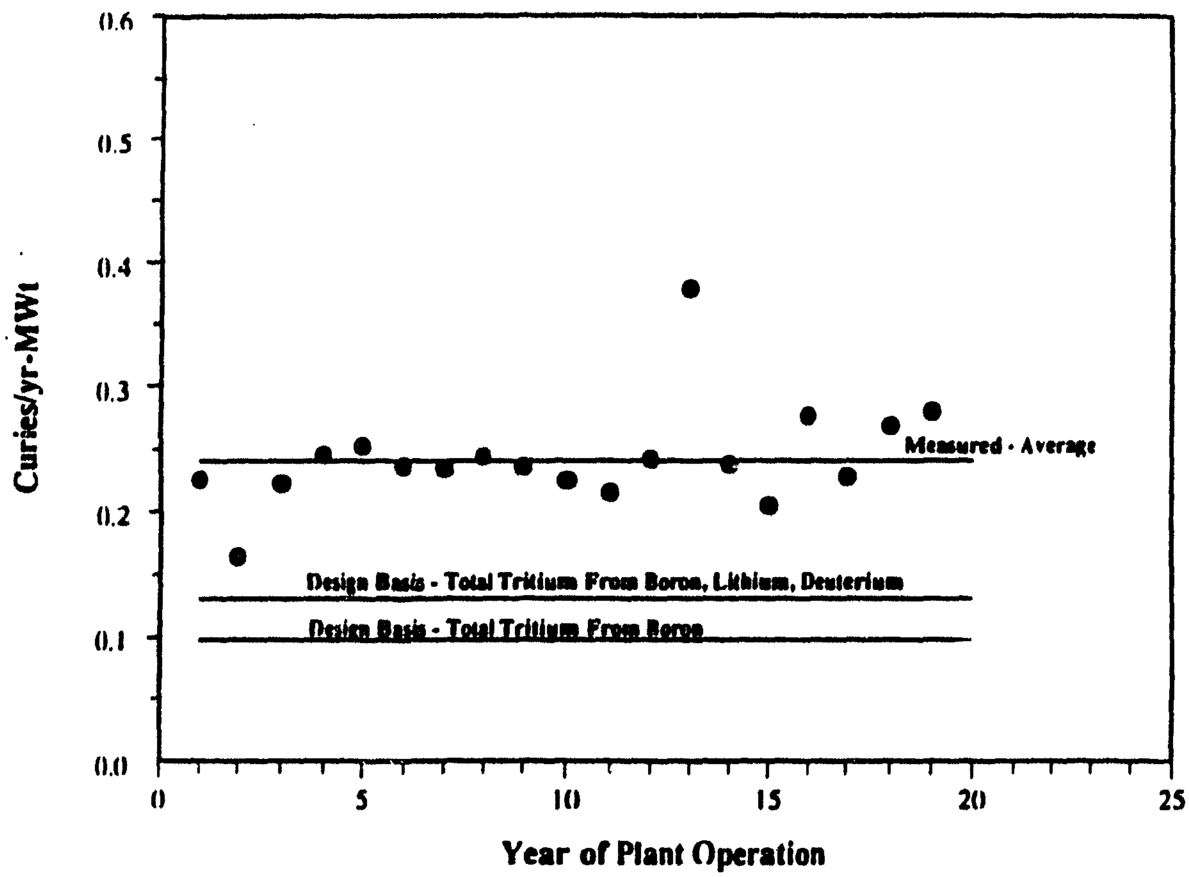

Figure 2.11.2-1
Average Annual Tritium Discharges from Operating Westinghouse Nuclear Plants

2.11.3 LIQUID WASTE MANAGEMENT SYSTEMS (SRP 11.2)

2.11.3.1 Acceptance Criteria

The acceptance criteria for the liquid radwaste treatment systems is that the system have the capability to meet the requirements specified in 10 CFR Part 20.106 - "Concentrations in effluents to unrestricted areas" (superseded by Subpart D of the revised 10 CFR 20 - "Radiation Dose Limits for Individual Members of the Public") as it relates to radioactivity in effluents to unrestricted areas, 10 CFR 50.34a as it relates to the design of waste management systems necessary to control releases of radioactive effluents to the environment, GDC 60 as it relates to waste management system design and the control of releases of radioactive materials to the environment, GDC 61 as it relates to radioactive waste systems design that assures adequate safety under normal and postulated accident conditions, and 10 CFR Part 50, Appendix I, as it relates to the numerical guides for dose design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion.

2.11.3.2 Evaluation

Background

The tritium in the primary coolant exists as tritiated water which cannot be removed by normal waste processing techniques (e.g., ion exchange, filtration, evaporation). Owing to its relatively long 12.3 year half-life, the tritium activity will eventually build up to excessive concentrations in the water volumes within the plant if not released from the plant. The buildup of tritium in plant liquid volumes can create undesirable radiological conditions if the concentration increases to levels on the order of 2-4 μCi/g of water. Control of tritium in nuclear stations through the release of tritiated water is the most practical means available, since tritium separation in not currently feasible from an economic standpoint and the radiological hazards associated with the release of tritium are minimal in that release concentrations and off-site doses can be maintained well below regulatory limits.

Any tritium that is released to or generated in the RCS tends to be distributed throughout the NSSS water volumes over time. During operation, RCS fluid is routed throughout various auxiliary systems such as the chemical and volume control system (CVCS), liquid radwaste system, and gaseous radwaste system. Water additions to the RCS for boron concentration dilution and as makeup for leakage during the cycle, define the rate of buildup of tritium in the RCS. Also, the primary coolant that is removed from the system by either boron dilution or leakage is generally discharged from the plant. During refuelings, tritiated reactor coolant is recirculated by the residual heat removal (RHR) system and mixed with the refueling cavity water, which is from the refueling water storage tank (RWST), and with the water in the spent fuel pit (SFP).

Operating commercial reactor plants eventually release all of the tritium that is introduced to the primary coolant to the environment. The major release pathway for the tritium is in liquid waste plant effluents. These releases are batch releases via a monitor tank that is sampled and analyzed prior to release into a dilution flow of non-radioactive water. This diluted flow is then routed to the environment (e.g. river, lake, ocean) where the concentration is further

reduced. The rate of release from the tank is controlled such that the diluted concentration does not exceed applicable regulatory and plant Technical Specification limits. The liquid effluent release rate from the monitor tank is controlled based on the activity concentrations in the tank and the flow rate of the dilution water. The amount of dilution flow is highly variable from plant to plant and can vary from about 3,000 gpm to 500,000 gpm. Although additional tritium input to the primary coolant system from TPBARs could have a significant negative impact on plants operating with low dilution flows in the 3,000 - 5,000 gpm range, most plants with cooling towers, including the reference plant, have dilution flows in the range of 15,000 - 30,000 gpm. Dilution flows of this magnitude are adequate for reducing the effluent concentration to acceptable levels.

Some operating plants process primary coolant in order to recover the soluble boron that exists in the reactor coolant system for core reactivity control. This re-use of this tritiated water, as opposed to batch release from the plant, would require a modification in plant operations in that some of the processed coolant that would normally be used as makeup water would have to be discharged. However, this is not expected to have a major impact on plant design or operability.

Currently, the tritium releases through the combined liquid and vapor pathways of PWRs, as presented in the plant Safety Analysis Reports (SARs), are 0.4 Ci/yr per MWt (NUREG-0017, Rev. 1, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors", April 1985). In the SARs, the quantity of tritium released through the liquid pathway is based on the calculated volume of liquid released, with a primary coolant concentration of 1.0 μCi/ml, up to a maximum of 90% of the total quantity of tritium calculated to be available for release. It is assumed that the remainder of the tritium produced is released as a gas from building ventilation exhaust systems. The release rate of 0.4 Ci/yr/MWt is based on a review of release rates from a number of PWRs and data from specific measurements of tritium inventory and releases from an operating 2-loop Westinghouse PWR. Operations with TPC can be expected to increase the amount of tritium within the plant as described in Section 2.11.2. Thus, plant liquid and gaseous tritium effluent releases can be expected to increase.

Tritium Control Considerations

As discussed in Section 2.11.2, a TPC increases the total annual tritium release to the coolant by a factor of approximately five. The RCS concentration increase is roughly proportional to the increased source such that the projected RCS concentration with TPBARs at the reference plant would increase from the current value of 1.6 - 2.0 μCi/gm to approximately 8 - 10 μCi/gm if no actions are taken.

Plant operation at a tritium concentration of this magnitude is undesirable, since the concentrations in the RCS, SFP, and RWST would build up to levels that would severely impact worker exposures. Also, worker efficiency would be reduced and costs would increase due to the need for additional personnel radiation protection measures, such as the use of plastic protective clothing and increased use of respirators and/or self-contained breathing apparatus gear. Thus, to avoid exceeding reasonable concentrations in the RCS, SFP, and RWST, the tritium must be removed from the RCS.

The two pathways for removal of the tritium activity are

1. dilution and discharge of tritiated water in liquid effluents, and

2. evaporation of tritiated waste and discharge in airborne effluents.

The preferred method of accommodating the increased tritium load is to continue the current practice of discharging the activity from the plant site via the liquid effluent pathway, particularly since discharge via the airborne pathway(s) would require additional equipment or equipment modifications. However, as noted above, the tritium inventory increases by a factor of five over that currently experienced. Thus, a much larger volume of primary coolant would have to be discharged as compared to current operating plant practices.

Liquid Releases for Boron Control

The amount of primary coolant that can be expected to be discharged as a result of feed and bleed operations related to control of boron in the RCS during a cycle is determined from:

$$C_f = C_i \times \exp(-F_B/V)$$

where:

C_f = final boron concentration, ppm

C_i = initial boron concentration, ppm

F_B = total volume processed for boron control, gallons

V = RCS volume, gallons

$$F_B = V \times \ln(C_i / C_f)$$

Critical boron concentrations for the TPC operating cycles are illustrated in Figure 2.11.3-1. In general, the values are similar to those for 18-month cycles for the reference plant. The dilution required over an equilibrium cycle for both the TPC and a typical conventional core is found to be approximately 5 RCS volumes over an 18-month fuel cycle, with the majority of the volume generated in the latter part of the fuel cycle when boron concentrations are lowest. A representative value for a typical year of plant operation is considered to be on the order of 2-3 RCS volumes per year.

Tritium Control by Liquid Discharge

An evaluation of the amount of liquid discharge that would be required to maintain reasonable tritium concentration levels in the reference plant was conducted using a computer model that simulates tritium production, release and mixing in various water volumes. Nine operating cycles were based on conventional cores; the tenth and subsequent cycles were based on the

TPC design. The sources of tritium in the primary coolant for the TPC operating cycles are summarized in Table 2.11.3-1.

Figure 2.11.3-2 illustrates the effect of RCS discharge rates on peak tritium concentrations in the RCS. The first 9 cycles of operation are based on an RCS discharge rate of 2 RCS volumes per year. This results in peak tritium concentrations of 1.5-2.0 μCi/g, which are consistent with the reference plant measured data. The discharge of 2 RCS volumes per year is also consistent with the estimated value of the amount of RCS discharged annually for boron control for the reference plant. After converting to TPC at cycle 10, assuming the RCS discharge remains at 2 volumes per year, the RCS activity increases to 8-9 μCi/g, or approximately 2 ½ times the recommended maximum of 3.5 μCi/g. If the amount of discharge is increased to 10 RCS volumes per year, the RCS concentration does not increase significantly above that associated with a conventional core. This is equivalent to an annual average flow of approximately 1.2 gpm.

The effects of RCS dilution are also illustrated in Figure 2.11.3-3. As shown in this figure, the amount of RCS discharge must be increased to about 6 RCS volumes per year in order to remain below the recommended maximum RCS tritium concentration of 3.5 μCi/g. Also, it is noted that even with this limited amount of RCS discharge, the spent fuel pit concentrations remain well below the recommended limit of 2.5 μCi/g. An RCS discharge volume of 6 RCS volumes per year is equivalent to an annual average of 0.7 gpm.

From an in-plant ALARA perspective, it would be preferred to operate with tritium concentrations below the recommended upper limit values of 3.5 μCi/gm in the primary coolant and 2.5 μCi/gm in the refueling cavity and the spent fuel pit water. As mentioned above, reactor coolant tritium concentrations in the range of 1.5 - 2.0 μCi/gm will be maintained if the amount of RCS liquid releases is increased to 10 system volumes per year. However, it should be noted that the above evaluation does not consider evaporative losses from the SFP and the cavity water during refueling. This would tend to decrease the required number of discharge volumes.

Impact of TPC Operation on Plant Discharge Limits

The release of the additional tritium inventory associated with TPC operation in plant effluents must not compromise the applicable release concentration and dose limits. Plant effluent releases are governed by the plant Technical Specifications. The reference plant Technical Specifications refer to the plant Offsite Dose Calculation Manual (ODCM), and the ODCM reflects the applicable 10 CFR 20 and 10 CFR 50, Appendix I criteria.

Figure 2.11.3-4 is a schematic diagram that illustrates the major plant structures, systems and components associated with a nuclear station. Plants may have a circulating water system that does not include the cooling towers included in the figure. Such plants generally have a dilution capability that is an order of magnitude greater than plants with cooling towers. As shown in this schematic, the two major discharge pathways are the "Plant Stack" for airborne effluents and "Blowdown to Plant Discharge" for the liquid effluents.

Discharge Concentration Limits

The effluent concentration limit specified by 10 CFR 20 for tritium must be considered in assessing the impact of the increase in tritium inventory in the RCS and the additional activity discharges that will be necessary to control an in-plant buildup. The effluent concentration limit in water from Appendix B of paragraphs 20.1001-20.2401 Table 2, column 2, of 10 CFR 20 is:

$$C^{lim} = 1 \times 10^{-3}\ \mu Ci/ml$$

However, the reference plant ODCM indicates that release of 10 times this 10 CFR 20 concentration is permitted based on an exemption granted by the NRC following issuance of the revised 10 CFR 20 in 1993, as allowed per 10 CFR 20. Similar exemptions have been granted to other PWR plants.

Since the reference plant has cooling towers and plants with cooling towers generally have relatively low dilution factors, the margin between actual discharge concentrations and the limits should be applicable (or bounding) for most operating plants. Based on the 15,500 gpm dilution flow associated with the reference plant and a discharge of 10 RCS volumes per year at a pre-dilution concentration of 2 μCi/gm , the concentration in the effluent stream is:

$$C^{dil} = C^{in} \times F_1 / F_2 = 2\ \mu Ci/gm \times (1.2\ gal/min) / (1.55 \times 10^4\ gal/min)$$

$$= 1.5 \times 10^{-4}\ \mu Ci/gm$$

This concentration is a factor of 6.7 less than the 10 CFR 20 limit and 67 less than the reference plant Technical Specification limit. However, this concentration is based on a discharge rate that is an "annual average" value and releases of liquid effluents are made in a batch mode. Thus, the margin below the discharge concentration limit must be weighed against the frequency and duration of the actual batch releases, where the instantaneous concentration limits of 10 CFR 20 and/or the plant Technical Specifications apply.

In order to substantiate the approach to discharge limits, recent tritium discharge data from the 1995 and 1996 Annual Radiological Effluent Reports for the reference plant were reviewed. The information from the reports is summarized in Table 2.11.3-2

From Table 2.11.3-2, the average annual discharge concentrations after dilution were in the range of 6×10^{-4} to 1×10^{-3} μCi/gm or about 6-10% of the applicable limit (i.e., plant Technical Specification limit of 10 CFR 20 effluent concentration limit increased by a factor of 10). Based on these reported concentrations, and considering the associated annual tritium effluent activities approached 1000 Ci/ unit, it follows that the limit would be reached with a total activity discharge on the order of 16,000 to 30,000 Ci per year; or much more than the projected discharge of less than 4300 Ci per year (see Table 2.11.2-3) when operating with a TPC. Thus, even if the projected increase in annual tritium release were to increase the annual average discharge concentrations, the expected margin is not reduced to the extent that the instantaneous concentration limits would be exceeded.

Dose to Members of the Public

Another criteria relative to plant activity releases that must be met is the ALARA requirement of 10 CFR 50, Appendix I (Reference 5), that is:

> ".. that radioactive material in effluents released from these facilities to unrestricted areas be kept as low as is reasonably achievable"

and for liquid discharged in particular,

> "the calculated annual total quantity of all radioactive material above background to be released from each light-water-cooled nuclear power reactor to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways of exposure in excess of 3 millirems to the total body or 10 millirems to any organ."

Relative to airborne releases, the reference plant ODCM stipulates that the dose to members of the public from I-131, I-133, tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents, from each reactor unit, to areas at and beyond the site boundary shall be limited to the following:

a) During any calendar quarter: Less than or equal to 7.5 mrem to any organ, and

b) During any calendar year: Less than or equal to 15 mrem to any organ.

The information in Table 2.11.3-2 can also be used in projecting the dose consequences to members of the public. From the table, the total dose from liquid effluents has historically been approximately 1% or 2 orders of magnitude below the applicable limits, and the total dose from airborne effluents has been more than 4 orders of magnitude below the applicable limits for gaseous effluents.

In order to assess the potential impact of TPC operation on the dose to the public, the actual dose information in Table 2.11.3-2 was increased by the ratio of the projected releases that include the additional activity inventory associated with a full complement of TPBARs (3378 Ci) and the actual plant tritium releases, i.e.,

$$D_{TPC} = D_{Actual} \times (I_{Actual} + I_{TPC}) \ / \ I_{Actual}$$

In addition, the relative contribution to the total dose from tritium releases, (i.e., about 15% of the total dose in liquid releases) was considered. The results are provided in Table 2.11.3-3. The projected dose, in terms of a percent of the ODCM limit, with TPBARs is generally less than 2% of the limit. That is, the fraction of the ODCM limit increases from approximately 0.01 (1%) with conventional core designs to about 0.02 (2%) with TPC operations.

It should be noted that with an increased discharge rate of primary coolant, there will be a shorter residence/decay time for nuclides in the liquid waste processing systems. Thus, an increase in the activity associated with short-lived isotopes can be anticipated. The increase of

activity in liquid effluents was evaluated and found to be insignificant (i.e., less than a 5% increase in total curies released). With this increase, the estimated annual releases remain a small fraction of the 10 CFR 50, Appendix I limit on liquid pathway releases.

Doses from airborne releases remain a negligible fraction of the applicable limit (i.e., <0.005% of the ODCM limit). That is, the fraction of the ODCM limit does not increase with TPC operation since the RCS, SFP and RWST concentrations remain unchanged. Since an additional amount of primary coolant is processed and released, more coolant will pass through the recycle holdup tanks. The vapor space of these tanks is routed to the waste gas system and eventually released in airborne effluents after holdup in the waste gas decay tanks. An increased amount of activity associated with short-lived nuclides could exist in the tanks. However, this inventory is small compared to that associated with input from the volume control tank, which can be continuously purged to the gas decay tanks and used to continuously remove gaseous activity from the primary system during normal operation, and during plant shutdowns and startups. Further, the waste gas system is designed to accommodate 1% fuel defects and has the capability for extended holdup times, in order to allow for sufficient decay of short-lived nuclides.

Impact of TPC Operation on Liquid Waste Management and Plant Operations

Since the discharge of liquid effluents are made in a batch mode, the increased amount of primary coolant that must be processed and discharged will have an impact on plant waste management and operations. This impact can be estimated based on information from the plant Effluent Release Reports.

The information in Table 2.11.3-2 was used in deriving the following conclusions relative to plant waste management:

a) The amount of liquid discharged was equivalent to 7-13 RCS volumes per year.

b) The average dilution factor was typically in the range of approximately 300 - 340.

c) The average concentration prior to dilution factor was typically 0.2 - 0.3 μCi/g.

d) Typical monitor tank discharge rates were 60-80 gpm.

Also, data from the Effluent Reports indicates that the number of batches discharged per year is generally in the range of 40 to 80 with an average tank release period of about 3 hours (typically 150 - 200 minutes).

The average amount of fluid discharged during a typical batch discharge is:

$$V = 175 \text{ min} \times 60 \text{ gpm} \approx 10{,}000 \text{ gallons} = 3.8 \times 10^7 \text{ g}$$

The reference plant liquid waste system design includes two monitor tanks at a volume of 5000 gallons each. Thus, the plant data indicates that the monitor tanks are full at the time of

discharge and, since an RCS mass is 2.3×10^8 g, the number of batches required to release an RCS volume is approximately:

$$N_{RCS} = 2.3 \times 10^8 \text{ g} / 3.8 \times 10^7 \text{ g} \approx 6$$

or 60 batch releases for 10 RCS volumes.

The actual amount of water discharged from the reference plant (i.e., 7-13 effective RCS volumes) is consistent with this value. However, this water volume is made up of primary coolant at a tritium concentration in the range of 0.5 to 2 µCi/g and water that is non-tritiated or at a low concentration, such as laundry and hot shower wastes, floor drains, secondary side water, etc.

$$V_{Release} = V_{RCS} + V_{non\text{-}RCS}$$

The fraction that is primary coolant can be estimated by considering the dilution of the RCS coolant tritium concentration. Considering an average discharge concentration prior to dilution of 0.25 µCi/g based on information in Table 2.11.3-2 and an average concentration of about 1 µCi/g in the coolant, the nominal RCS discharge is:

$$R_{RCS} = 10 \times 0.25 \ \mu\text{Ci/g} \ / \ 1 \ \mu\text{Ci/g} = 2.5 \text{ RCS volumes per year}$$

This is considered to be a reasonable estimate in that this is about the amount of RCS that is removed by feed-and-bleed for RCS boron control, and it is also consistent with the observed peak tritium concentrations in the primary coolant of 1.5-2 µCi/g.

Since 10 RCS volumes should be discharged in order to limit tritium buildup in the RCS, operations with TPC will require that an additional 7.5 volumes of primary coolant be discharged. Based on the assumption that the non-tritiated effluent volume remains constant, the total liquid effluent associated with TPC operation is then:

$$R_{TPC} = 2.5 + 7.5 + (10\text{-}2.5) = 17.5 \text{ RCS volumes}$$

and the projected batch discharge frequency with a TPC would increase from about 60 releases (average of 40-80 in 1995 and 1996) to:

$$N_{TPC} = N_{RCS} \times R_{TPC}$$

$$= 6 \text{ batch releases per RCS volume} \times 17.5 \text{ RCS volumes}$$

$$= 105 \text{ batch releases}$$

or about a 75% increase in the number of releases.

In terms of time of release, this converts to:

T_{TPC} = 105 releases x 175 minutes per batch

$= 1.84 \times 10^4$ min = 306 hours per year

or an increase over current batch release time of approximately:

dT = (105 - 60) x 175 minutes per batch

$= 7.9 \times 10^3$ min = 130 hours per year

Evaluation of TPBAR Failures

As discussed in Section 2.11.2, a potential tritium release scenario is the failure of two TPBARs, such that the inventory of the TPBARs is released to the primary coolant. The occurrence of two failed TPBARs is considered to be beyond that associated with reasonable design basis considerations. However, the impact of the two failed rods will be addressed in order to assess the impact of a highly abnormal situation on plant radiological and waste management issues.

The inventory in two failed TPBARs was determined (Section 2.11.2.2) to be 20,000 Ci per cycle. It is considered reasonable to expect that most of the activity would be retained in the getter material. However, in lieu of specific release rate information an immediate release of tritium from the TPBARs is conservatively assumed.

Since the tritium release from TPBARs by permeation mechanisms could vary from a small added amount to the design basis value, an observed tritium production increase may not readily be differentiated from TPBAR failure. However a failure or failures of the TPBAR cladding could be indicated by relatively rapid increases in the tritium RCS concentrations, particularly if the magnitude is greater than that associated with the design basis value associated with permeation. In order to properly track and manage any unanticipated events, such as TPBAR failure, it will be necessary to develop appropriate procedures and action plans. Such procedural and administrative measures will serve to:

- ensure that the TPC is consistent with design objectives
- serve as a trigger for increased data monitoring and trending
- initiate recovery actions
- aid in the development of appropriate actions for minimizing the impact of unexpected tritium production increases on:
  - worker dose
  - dose to members of the public
  - low level waste

Assuming the worst case scenario of the design basis TPC tritium sources coincident with two failed rods, the projected impact on plant radiological and waste management issues is discussed below.

Given that both of the rods fail at the same time, there are two bounding assumptions that can be made in the release scenario, i.e.

1. failure occurs at the beginning of the cycle (BOL)
2. failure occurs at the end of the cycle (EOL)

Failure at BOL

If the failures occur at BOL, the tritium inventory will be released to the primary coolant throughout the cycle as it is generated in the TPBARs. The associated tritium releases for the failed rod scenario (I_{Fail}) would be additive to the postulated releases associated with the permeation of tritium through the getter and TPBAR cladding (L_{TPC}). Thus, the total additional tritium inventory that could be released is:

$$R_{\Delta} = L_{TPC} + I_{Fail} = 3378 + 20{,}000 \text{ Ci/cycle} \times (365 \text{ days/year} / 494 \text{ days/cycle})$$

$$= 3{,}378 + 14{,}777 = 18{,}155 \text{ Ci/yr}$$

and when added to the expected inventory from the sources not associated with TPC (i.e., 890 Ci/year), the total annual production is 19,045 Ci/year.

It was determined that on the order of 30 RCS volumes per year of primary coolant would have to be discharged in order to accommodate a tritium production rate in the RCS of 20,000 Ci/yr and still remain below the recommended RCS concentration limit of 3.5 µCi/g. The current annual releases at the reference plant were estimated to average about 2.5 system volumes of primary coolant and roughly 7.5 system volumes of non-tritiated water. Hence, the estimated discharges would have to be increased from 10 system volumes to nearly 40 system volumes. Based on past experience at the reference plant, it is estimated that the discharge of one system volume involves 6 batch releases. Thus, the release of 40 system volumes would require an average discharge frequency of roughly 240 batches per year, or one batch every 1-2 days. Further, the duration of the discharge may have to be increased from the current average of about 3 hours per batch in order to meet discharge concentration limits, due to the fact that the concentration of tritium in the waste liquid would be higher than currently experienced.

The average annual discharge concentrations after dilution with conventional core designs (see "Concentration after Dilution" in Table 2.11.3-2) were about 6-10% of the applicable limit, as discussed earlier in this section. Based on these reported concentrations and considering the associated annual tritium effluent activities approached 1000 Ci/ unit, it follows that the limit would be reached with a total activity discharge on the order of 16,000 to 30,000 Ci per year, which is the magnitude of the projected activity release scenario.

Since liquid discharges are made in a batch mode, consider the discharge frequency and duration that would be required to accommodate the increased source term.

The current average dilution factor experienced at the reference plant is about 300-350. Assuming that in this off-normal condition, the RCS tritium concentration is maintained at the recommended limit of 3.5 μCi/g, the discharge concentration after dilution is:

$$C_d = 3.5\ \mu Ci/g\ /\ 325 = 0.011\ \mu Ci/g$$

which is approximately the same as the plant Technical Specification discharge concentration limit of 0.01 μCi/g. However, in practice, the cooling tower flow can be higher by a factor of two and the discharge flow can be adjusted such that additional margin is available between the discharge concentration and the concentration limit. In addition such measures as pre-dilution of the primary coolant waste water with non-tritiated secondary side water, and/or evaporation and discharge via the plant vent are potential remedial actions.

The impacts on doses to the public are evaluated below.

Liquid Releases - As indicated above, the projected dose impact of the design basis TPC release of roughly 4000 - 4300 Ci/year was found to be an increase in the fraction of the ODCM limit from approximately 0.01 (1%) with conventional core designs to about 0.02 (2%). Conservatively neglecting the contribution of nuclides other than tritium, the projected impact of a source of 20,000 Ci/yr is to increase the fraction of the limit by the ratio of the sources. Thus the fraction of the limit would be increased to:

$$F_{limit} = 0.02 \times 20,000\ Ci/yr\ /\ 4,000\ Ci/yr = 0.10$$

or 10% of the ODCM limit.

Airborne Releases - The fraction of the ODCM limit for airborne releases is conservatively estimated in Section 2.11.4.2 to increase from less than 0.005% of the ODCM limits with the design basis TPC sources to less than 0.1% of the ODCM limit with the increased tritium from 2 failed TPBARs. Since this release pathway is such a small fraction of the applicable limit, releases in airborne effluents could be considered as a means of reducing dose and or augmenting the systems and equipment associated with the processing and discharge of primary coolant through the liquid pathway (e.g., evaporator steam routing, forced evaporation of tank volumes, etc.). Releases via the airborne pathway are considered to be a viable alternative, based on past experience at Three Mile Island and other LWR plants which have limitations on liquid effluents.

In summary, the projected impacts of the tritium releases associated with the design basis TPC sources, as well as continued operation with two failed TPBARs from BOL are:

a) an increase of concentrations in the primary coolant to the recommended limit,

b) increased liquid effluent discharges to approximately 40 RCS volumes,

c) releases at the maximum dilution flow and/or a reduction in monitor tank discharge flow rate in order to meet discharge concentration limits,

d) increases in dose to members of the public are projected; the total dose projected is approximately 10% of the applicable limit for liquid effluents and 0.1% of the airborne limit, and

e) the possibility of the promotion of release via the airborne pathway for which there is additional margin in applicable limits.

Failure at EOL

If it is assumed that the failure of TPBARs occurs at or near the end of the cycle, and conservatively assuming that the complete tritium inventory of two TPBARs is immediately released to the primary coolant, a rapid increase in the existing RCS tritium concentration would be observed. The RCS tritium concentration would increase from an existing level assumed to be 2μCi/g to:

$$C_o = (I_{TPBARs} + I_{RCS}) \,/\, M_{RCS} = (20{,}000 + 460)\ \text{Ci} \times 10^6\ \mu\text{Ci/Ci} \,/\, 2.3 \times 10^8\ \text{g}$$

$$= 89\ \mu\text{Ci/g}$$

An increased RCS tritium concentration of this magnitude would not necessarily preclude continued plant operation. However, it may severely limit or prohibit access inside the containment at power, depending on the level of primary coolant leakage. The recovery from the TPBAR failures would be to reduce the concentration to the maximum recommended concentration (i.e., 3.5 μCi/g) in a timely manner and, in particular, prior to flooding the cavity prior to refueling. If this activity is not removed prior to filling the refueling cavity, it would result in increased worker exposure and interference with refueling operations. The amount of diverted letdown (feed-and-bleed) required to accomplish a reduction to 3.5 μCi/g can be estimated from:

$$C = C_o \exp(-R\,T)$$

where:

$$R = \text{effective removal constant} = \dot{Q} \,/\, V$$

which reduces to:

$$R\,T = \ln (C_o / C)$$

and for:

$$C_o / C = 89 / 3.5 = 25$$

$$\ln (C_o / C) = \ln (25) = 3.2$$

The time to reduce the concentration at a letdown (feed-and-bleed) flow of 120 gpm is:

$$T = \ln(C_o / C) / (Q/V) = 1600 \text{ min}$$
$$= 27 \text{ hours}$$

which is considered to be reasonable for recovery for a highly abnormal situation.

From the above, it is noted that the "recovery time" , T, is inversely proportional to the diverted letdown flow, so in the event that tank capacity or processing capability limits the feed-and-bleed flow rate a recovery time in excess of 27 hours would be required.

Summarizing, the projected impacts of the tritium releases associated with the design basis TPC sources coincident with two failed TPBARs at the EOL are:

a) a potential rapid increase in the RCS tritium concentration to a value that could be 89 μCi/g, which could limit or temporarily restrict access inside the containment until the tritium concentrations are reduced to acceptable levels,

b) increased feed-and-bleed processing or temporary storage with subsequent processing of primary coolant to reduce levels to at least the maximum recommended concentration of 3.5 μCi/g,

c) releases at the maximum dilution flow and/or a reduction in monitor tank discharge flow rate in order to meet discharge concentration limits,

d) increases in dose to members of the public; total projected dose is approximately 10% of the applicable limit for liquid effluents and 0.1% of the airborne limit, and

e) the possibility of the promotion of release via the airborne pathway for which there is additional margin in applicable limits.

2.11.3.3 Conclusions

The impact of operation with a TPC is expected to have a negligible impact on the design basis and realistic fission and corrosion product sources that are considered in the determination of plant releases and the waste management systems used in the control of these releases.

However, design basis tritium sources with a TPC are expected to increase the total amount of tritium that is generated in the plant by a factor of five (i.e., from about 890 curies per year to approximately 4268 curies per year). The method of accommodating the increased tritium inventory is to continue the current practice of discharging the activity in the liquid effluent pathway. This will limit the concentration buildup within the plant and minimize the radiation exposures to plant workers with a minimal impact on applicable regulatory requirements. That is, the tritium discharge concentrations can be maintained to less than 10% of the reference plant Technical Specification limit, and the fraction of the plant's Offsite Dose Calculation Manual (ODCM) limit attributed to tritium in liquid discharges is projected to increase from approximately 1% with conventional core designs to only about 2% with a TPC. Because the primary coolant tritium levels are not changing, the doses in airborne effluents are not expected to be affected, and would remain well below (e.g., less than 0.005%) of the ODCM dose limit.

In addition to the potential permeation of tritium through the TPBARs, a postulated tritium release scenario involving the failure of two TPBARs was evaluated. Such a failure is considered to be beyond that associated with the design basis sources for TPBAR permeation. This was addressed in order to assess the impact of this abnormal situation on plant radiological and waste management issues.

The EOL failure of two TPBARs results in a potential total tritium release of 20,000 curies to the primary coolant using the design basis TPC tritium production value. In the unlikely event that this would occur, substantial increases in intentional activity releases would be required in order to maintain reasonable in-plant activity concentrations. Although this could impact normal waste management operations and procedures, plant operation within prescribed regulatory limits is viable, with projected doses that are less than 10% of the applicable liquid effluent limits and less than 0.1% of airborne release limits. Plant procedures that specify action levels and recovery methods would be in effect should TPBAR failures occur.

It should be noted that the simultaneous and immediate inventory release of two TPBARs is highly conservative. As noted in Section 3.7.3,

1. The failure of two TPBARs is based on a conservative projection of burnable absorber rod experience, i.e., there is no evidence of WABA failures or deficiencies.
2. The assumption of immediate release of tritium from the TPBARs is conservative since release of tritium from a failed TPBAR requires the tritium to diffuse from the lithium aluminate pellets and getters into the fluid, through narrow gaps inside the rods, and through the breach to the primary coolant.
3. The leaking TPBARs are considered to be waterlogged, and the diffusion transport in liquids is a slow process that should delay tritium release from leaking TPBARs.

Thus, based on operating plant experience with other types of burnable poison rods, the failure of two rods is highly conservative, and the rate of release is not expected to be immediate. These factors would result in a reduced impact (with respect to the worst-case predictions) on plant waste management and recovery methods.

In conclusion, whether the plant is experiencing the design basis TPBAR permeation of tritium, or a situation involving two failed TPBARs, the reference plant will continue to meet the applicable release concentration and dose limits as provided in the plant's Offsite Dose Calculation Manual.

Table 2.11.3-1 Tritium in the Primary Coolant with the TPC		
Source	Initial Cycle Ci	Equilibrium Cycle Ci
Fuel and TPBARs	5470	5450
BP	31	35
Boron	661	711
Lithium	162	161
Deuterium	4	4
Total	6328	6361

Table 2.11.3-2
Reference Plant Effluent Release Data
Reported Values

		Year of Release	
		1995	**1996**
H_3 Release, Ci			
Unit 1	Liquid	554	802
	Gas	216	92
	Total	770	894
Unit 2	Liquid	415	834
	Gas	145	81
	Total	559	915
Units 1 & 2		1329	1809
Volume Released, liters			
Unit 1		2.6E+06	3.4E+06
Unit 2		1.6E+06	3.0E+06
Units 1 & 2		4.2E+06	6.4E+06
Volume of Dilution Water, liters			
Unit 1		7.5E+08	1.2E+09
Unit 2		5.4E+08	1.0E+09
Units 1 & 2		1.3E+09	2.2E+09
Concentration after Dilution, (μCi/g)			
Unit 1		7.4E-04	6.0E-04
Unit 2		9.8E-04	9.8E-04
Units 1 & 2		7.9E-04	7.3E-04
No. of Releases (Typical average release time = 150-200 minutes)			
Unit 1		56	78
Unit 2		40	58
Units 1 & 2		96	136
Liquid Release - % of ODCM Limit (Total Body < 3 millirem/yr)			
Unit 1		1.2	0.8
Unit 2		1.0	1.0
Iodine and Tritium Airborne Release - % of ODCM Limit (Total Body < 15 millirem/yr)			
Unit 1		0.004	0.003
Unit 2		0.002	0.003

Table 2.11.3-3 **Projected Impact of TPC on Reference Plant Effluent Dose to a Member of the Public**		
	Year of Release	
	1995	**1996**
Projected Release with TPC, Ci		
Unit 1	4148	4272
Unit 2	3937	4293
Liquid Release - Projected % of ODCM Limit		
Unit 1	2.0	1.3
Unit 2	1.9	1.5

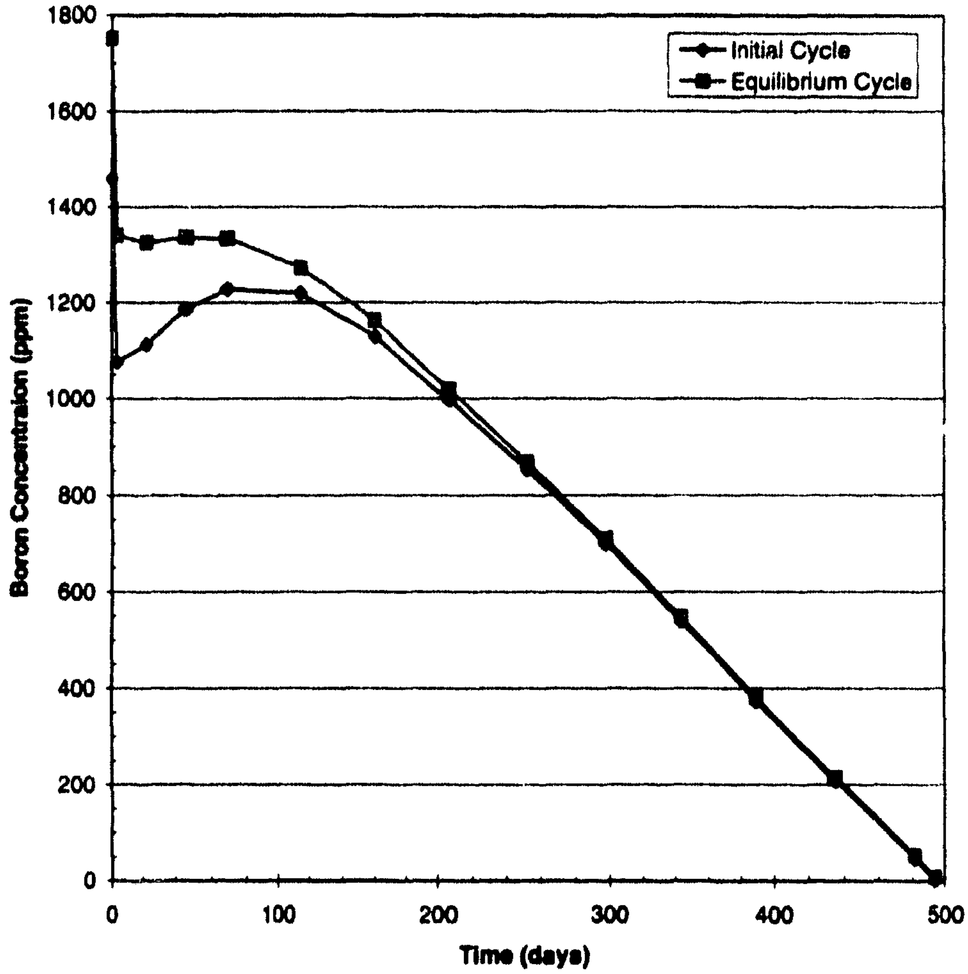

Figure 2.11.3-1
TPC Critical Boron Concentrations

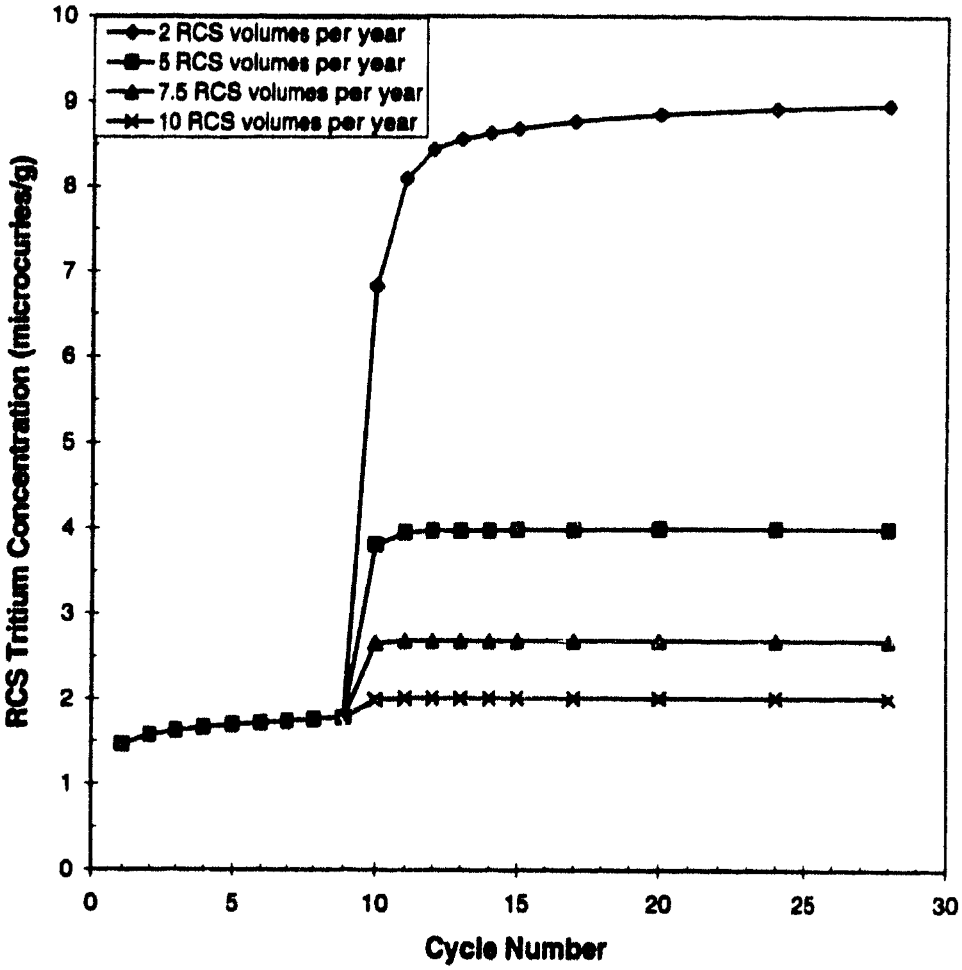

Figure 2.11.3-2
Effect of Conversion to TPC on Peak Tritium Concentration in the RCS
For the Reference Plant

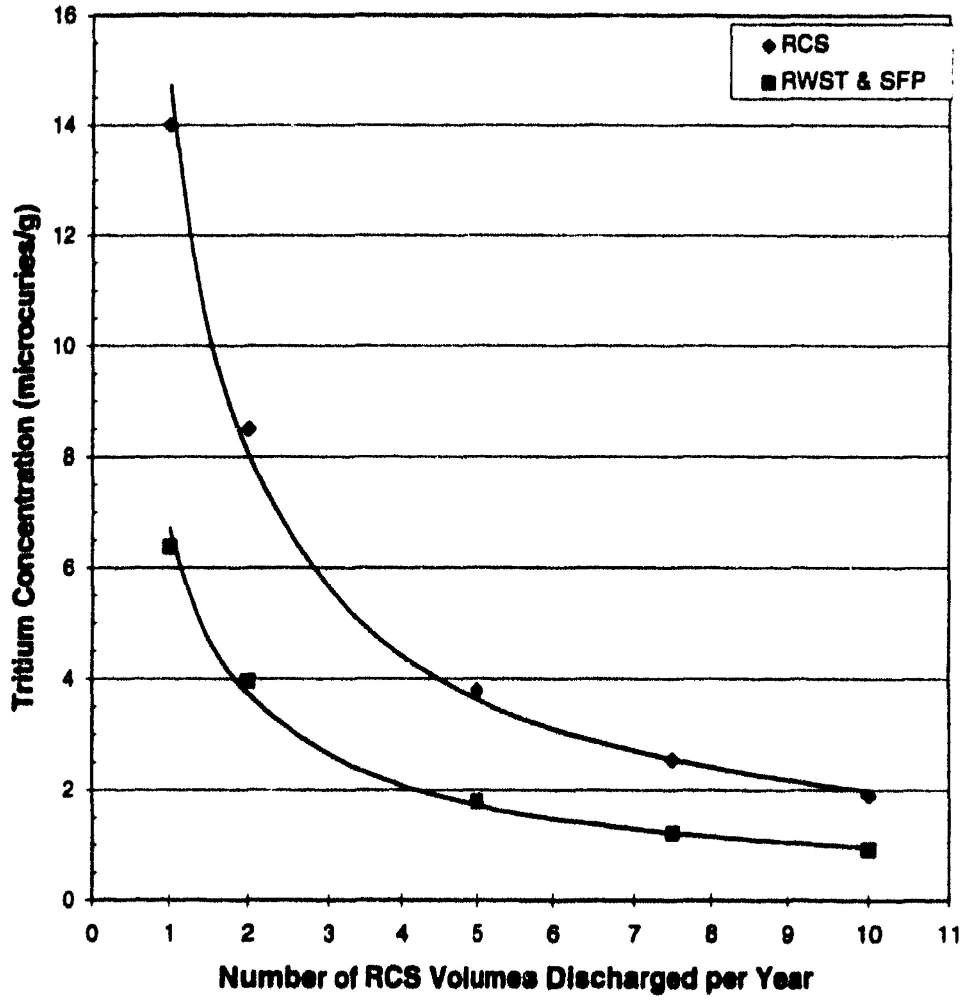

Figure 2.11.3-3
Peak Tritium Concentrations in the RCS, RWST, and SFP as a Function of RCS Discharge Rates

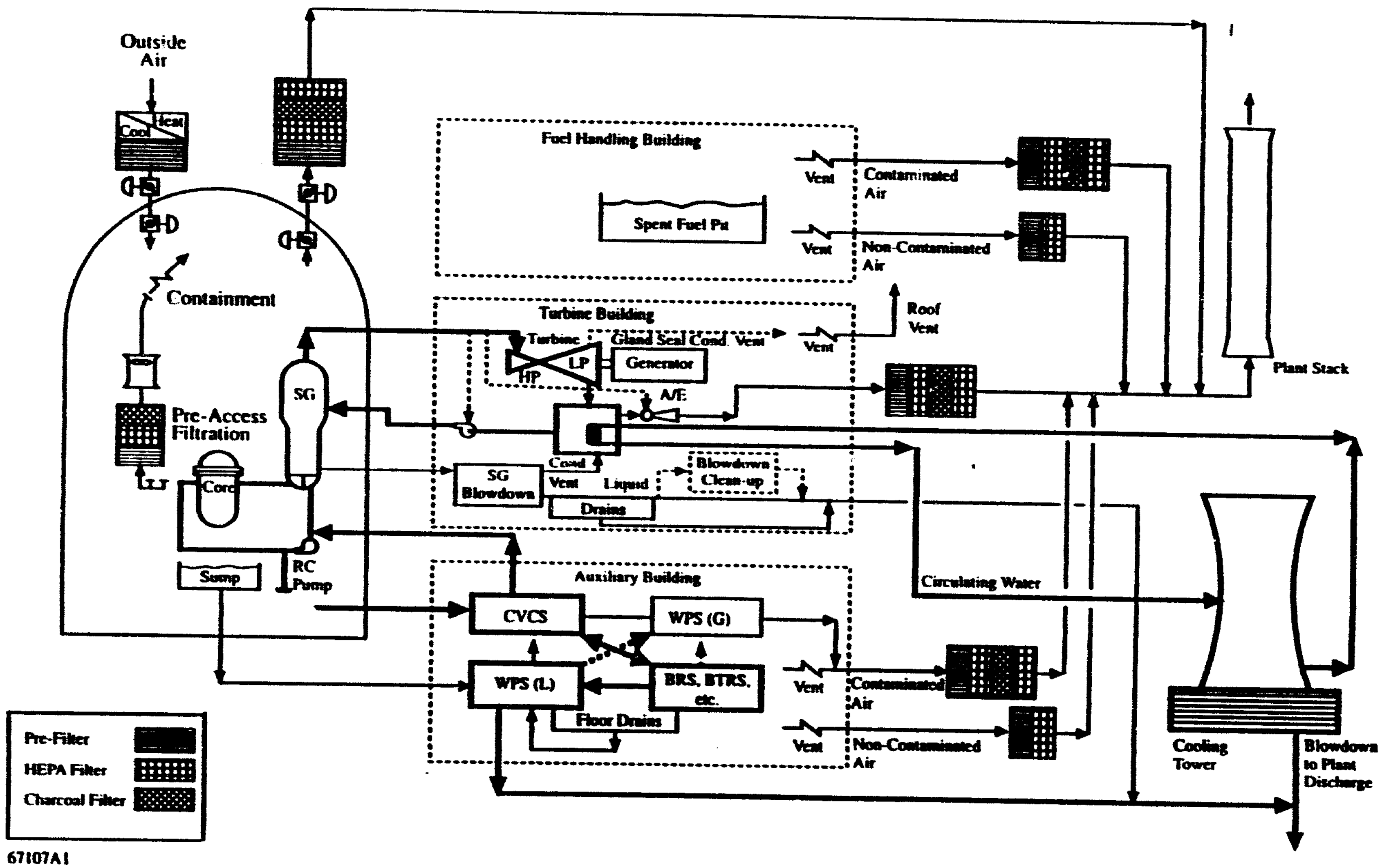

Figure 2.11.3-4
Schematic Diagram of Major Plant Structures, Systems, and Components

2.11.4 GASEOUS WASTE MANAGEMENT SYSTEMS (SRP 11.3)

2.11.4.1 Acceptance Criteria

The acceptance criteria for the gaseous waste management systems is that the system meet the relevant requirements specified in 10 CFR Part 20.106 - "Concentrations in effluents to unrestricted areas" (superseded by Subpart D of the revised 10 CFR20 - "Radiation Dose Limits for Individual Members of the Public") as it relates to radioactivity in effluents to unrestricted areas, 10 CFR50.34a as it relates to equipment necessary to control releases of radioactive effluents to the environment, GDC 3 as it relates to protection of the gaseous waste handling and treatment systems from the effects of an explosive mixture of hydrogen and oxygen, GDC 60 as it relates to waste management system design and the control of releases of radioactive materials to the environment, GDC 61 as it relates to radioactive control in gaseous waste management systems and ventilation systems associated with fuel storage and handling areas, and 10 CFR Part 50, Appendix I, as it relates to the numerical guides for dose design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion.

2.11.4.2 Evaluation

As described in Section 2.11.3, the preferred method of accommodating an increased tritium inventory from TPC is to continue the current practice of discharging the activity from the plant site via the liquid effluent pathway. This approach is preferred, particularly since discharge via the airborne pathway(s) could require additional equipment and/or equipment modifications. However, in addition to the intentional release of tritium in liquid effluents, there is also normal evaporative losses from the refueling cavity water and the spent fuel pit. Since the control strategy will not result in significant change to the tritium concentrations in these water volumes, there should be no change between operations with TPC versus a conventional core.

However, even if it is conservatively assumed that the doses increase in proportion to a postulated increase in RCS activity from a typical operating range of 0.5 - 2 $\mu Ci/g$ to the maximum recommended concentration of 3.5 $\mu Ci/g$, the doses from airborne tritium releases remain a negligible fraction of the applicable limit. The fraction of the ODCM limit is projected to increase from less than 0.00005 (0.005%) to less than 0.1% with TPC operations.

The TPC has no additional impact on plant waste gas processing systems since the tritium exists as tritiated water (i.e., HTO or T_2O). The small additional load on the waste gas system equipment and/or equipment design is well within the system capabilities, which are based on 1% fuel defects.

2.11.4.3 Conclusion

The control strategy for controlling the buildup of tritium does not include use of the waste gas system(s). Therefore there should be no change between operations with TPC versus a conventional core.

2.11.5 SOLID WASTE MANAGEMENT SYSTEMS (SRP 11.4) EVALUATION

2.11.5.1 Acceptance Criteria

The acceptance criteria for the solid waste treatment system design is that the system meet the relevant requirements specified in 10 CFR Part 20.106 as it relates to radioactivity in effluents to unrestricted areas, 10 CFR 50.34a as it relates to equipment necessary to control releases of radioactive effluents to the environment, GDC 60 as it relates to waste management system design and the control of releases of radioactive materials to the environment, GDC 63 and 64 as they relate to the radioactive waste system design for monitoring radiation levels and leakage, and 10 CFR Part 71 as it relates to radioactive material packaging.

2.11.5.2 Evaluation

The amount of tritium in solid wastes is normally low and not expected to be impacted by operations with TPBARs. However, if additional liquid releases are necessary in order to control the buildup of tritium within the plant, ion exchange resin and filter usage may be increased in order to accommodate such plant releases. That is, the additional water that is released in plant effluents may require ion exchange and filtration in order to reduce the radioactivity and/or contaminant concentrations prior to discharge; thereby increasing the amount of associated solid waste.

Low Level Waste Volume

As described above, the use of the TPBARs could increase the normal RCS tritium concentration. Based on the design basis release from the TPBARs, on the order of 7½ reactor system volumes (about 750,000 gallons) will be discharged in order to maintain the in-plant tritium concentrations to acceptable levels. The path for diverting the reactor coolant for processing and release is described below.

Letdown is periodically diverted to the Boron Recycle System when a change to the RCS boron concentration is required. Makeup is then provided as a concentrated boric acid solution (boration operation) or as reactor makeup water (dilution operation). Assuming the plant remains at a constant power level, a gradual dilution operation is performed to reduce the RCS boron concentration at a rate of a few ppm boron per day, which generates waste (diverted letdown) for processing in the Boron Recycle System. In addition to this normal liquid waste, the additional waste due to TPBARs must also be processed.

The Boron Recycle System design is shown in Figure 2.11.5-1. Most operating plants have one of the two design arrangements. Arrangement 1 has a mixed bed upstream of the recycle holdup tank, but does not have a monitor tank. Arrangement 2 has a mixed bed downstream of the recycle holdup tank and does include the monitor tank

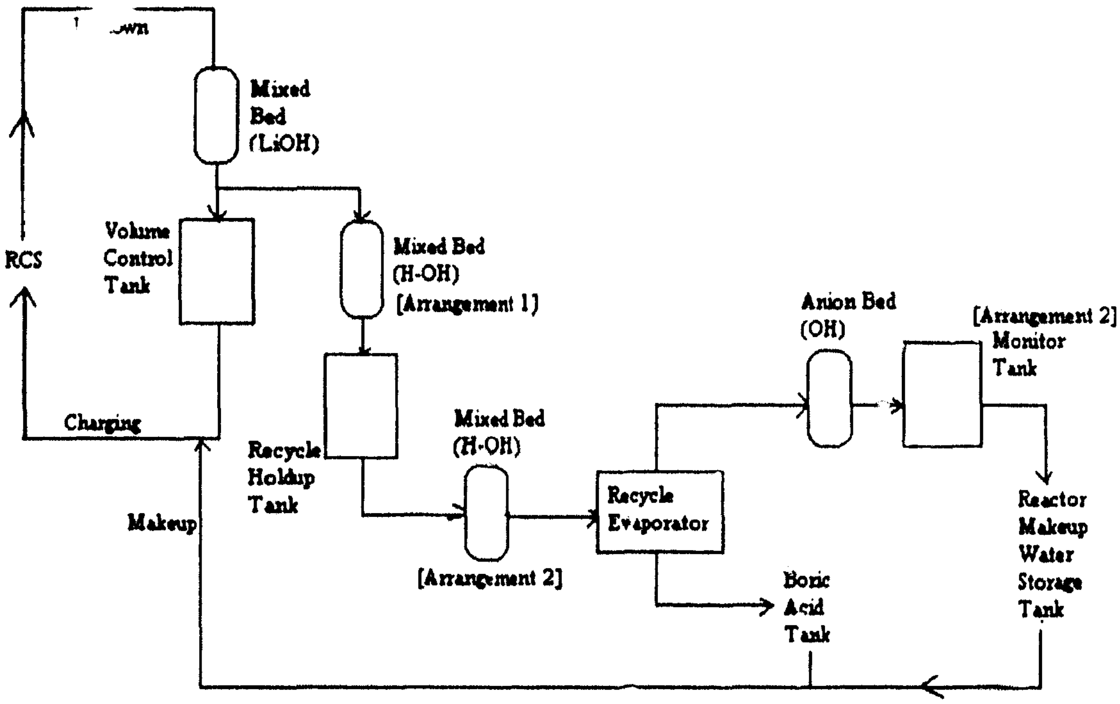

Figure 2.11.5-1
Schematic Diagram of Boron Recycle System

The additional waste from the use of TPBARs will increase the load on the demineralizer upstream of the evaporator. The two ions of highest concentration in the RCS are boron (anion) in the range of 0-1500 ppm and lithium (cation) in the range of 0.7 to 2.2 ppm. The anion portion of the mixed bed will quickly saturate with boron and the boron will subsequently pass through the bed. Lithium will take significantly longer to saturate the cation portion of the bed, since its concentration is much lower. When it is saturated, the cation resin's capacity to remove cesium will be reduced. Since cesium removal is highly desirable from a dose reduction point of view, it will be assumed that the resin bed will be changed when it is saturated with lithium.

Based on vendor resin capacity data, it has been estimated that the resin bed can hold approximately 3240 grams of lithium. The additional waste volume of 7 ½ system volumes contains approximately 3450 grams of lithium, at an average concentration of 1.5 ppm. Therefore, the additional number of resin bed changes due to the use of TPBARs is approximately one per year. For the bed volume of 30 ft^3, this is about a 10% increase in the estimated annual quantity of primary spent resins for the reference plant (i.e., 516 ft^3 for 2 units or 258 ft^3 per plant) and represents a small percentage of the total annual radwaste generation (Reference 1).

Low Level Waste Activity

The additional low level waste activity associated with TPC operation is estimated to be less than a 30 % increase in the current solid waste activity generation rate associated with the candidate plants. This estimate is based on the following data and assumptions:

- The total annual activity associated with current plant solid waste shipments is approximately 2000 Ci/yr (Reference 1, Table 11.4.2-5).
- On the order of three times the amount of liquid will be processed in the boron recycle and/or liquid radwaste systems in a plant with TPBARs as compared to a conventional plant.
- At least 90% of the total solid waste activity is associated with the CVCS system and is independent of the activity accumulated on boron recycle and waste systems filters and demineralizers that would be used in processing plant effluents.

Then, conservatively assuming that 10% of the total solid waste activity is increased in proportion to the amount of liquid that is processed:

$$\text{Activity increase} = 2000 \text{ Ci} \times 0.10 \times 3 = 600 \text{ Ci}$$

Thus, the estimated solid waste activity increase is from 2000 Ci to 2600 Ci which constitutes a 30% increase in the low level solid waste activity generation rate. Note that a review of operating data indicates that the value of 2000 Ci is conservative relative to actual annual solid waste activities, which are typically about 1000 Ci or less (Reference 2). However, the percentage increase in solid waste activity remains at 30%, based on the other parameters and assumptions used in the evaluation.

2.11.5.3 Conclusion

The additional number of resin bed changes due to the use of TPBARs is approximately one per year, or about a 10% increase in the estimated annual volume of primary resins and a small percentage of the total annual radwaste generation. The estimated solid waste activity increase is from 2000 Ci to 2600 Ci which constitutes a 30% increase in the low level solid waste generation rate. Consideration of two failed TPBARs results in an increased number of resin bed changes from one to approximately four per year.

2.11.5.4 References

1. Reference Plant - Final Safety Analysis Report Update, Section 11.4.
2. NUREG/CR-2907, "Radioactive Materials Released from Nuclear Power Plants", prepared for the USNRC by the Brookhaven National Laboratory.

2.11.6 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS (SRP 11.5) EVALUATION

2.11.6.1 Acceptance Criteria

The acceptance criteria for the process and effluent radiological monitoring instrumentation and sampling systems are based on meeting the relevant requirements specified in 10 CFR Part 20.106 - "Concentrations in effluents to unrestricted areas" (superseded by Subpart D of the revised 10 CFR 20 - "Radiation Dose Limits for Individual Members of the Public") as it relates to radioactivity in effluents to unrestricted areas, General Design Criteria 60 as it relates to waste management system design and the control of releases of radioactive materials to the environment, and General Design Criteria 63 and 64 as they relate to the radioactive waste system design for monitoring radiation levels and leakage.

2.11.6.2 Evaluation

Operating a plant with a TPC can be expected to increase the amount of tritium released to the RCS and plant tritium releases are expected to increase. The current process and effluent radiological monitoring instrumentation and sampling systems that are in place at the reference plant, as well as at other operating PWR plants, include the capability for monitoring the tritium levels within the plant and in plant effluent pathways. Although sample and analysis frequencies are expected to increase, new or additional process and effluent radiological monitoring and sampling systems or equipment are not anticipated.

2.11.6.3 Conclusion

Existing effluent radiological monitoring and sampling systems and equipment are adequate for use when the plant is operated with a TPC in order to assure that the radioactivity in effluents to unrestricted areas meets the release limits (10 CFR 20) and the General Design Criteria relating to waste management system design, control of releases of radioactive materials to the environment, and for monitoring of radiation levels and leakage.

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2.12 RADIATION PROTECTION

2.12.1 INTRODUCTION

The sections of Chapter 12 of the SRP are related to the plant programs and design features which are intended to maintain radiation exposures as low as is reasonably achievable. Section 2.12.2, below, describes the impact of TPBARs on radiation sources. The radiation protection features and dose assessments are discussed in Section 2.12.3, and the impact of TPBARs on the Operational Radiation Protection (Health Physics) Program is described in Section 2.12.4.

SRP 12.1
Assuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable (ALARA): The acceptance criteria for the SAR information associated with assuring that the Occupational Radiation Exposure is ALARA are based on meeting the relevant requirements specified in 10 CFR Part 50.34 and assuring that the SAR contains sufficient information related to the relevant requirements of 10 CFR 19 and 10 CFR 20. The pertinent section of 10 CFR 19 is paragraph 19.12 - "Instructions to Workers" as it relates to workers being informed of radiation sources, risks associated with radiation exposures, precautions and procedures to reduce exposure, and purpose and function of protective devices. The pertinent Section of 10 CFR 20 is paragraph 20.1(c) - "Purpose" as it relates to radioactivity in effluents to persons involved in maintaining radiation exposures As Low As Is Reasonably Achievable. The applicable section of 10 CFR 20, i.e., paragraph 20.1(c) - "Purpose", is superseded by the revised 10 CFR 20.1101 - "Radiation Protection Programs". The major focus in this section of a typical SAR is on demonstrating that an ALARA policy is in place, that there is a high level of management commitment to the ALARA policy, and that there is an organizational structure in place which facilitates the interaction of the radiation protection organization with design review groups. This information is not impacted by operation with a TPC. Other aspects of ALARA are considered in the Section 2.12.2 through 2.12.4.

SRP 12.2
Radiation Sources: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 20.101, 20.103, 20.104, 20.106, 20.207; 10 CFR 50.34; and 10 CFR 50, Appendix A, GDC 61. They deal with limiting radiation doses to protect individuals in restricted areas from whole or partial body exposures; limiting average concentrations of airborne radioactive materials to protect individuals in restricted areas and controlling inhalation or absorption of such materials; limiting exposure of minors to 1/10 of limits for adults; the determination of radiation levels and radioactive materials concentrations within the components of waste treatment systems; securing licensed materials against unauthorized removal; and fuel storage and handling and radioactivity control. An evaluation of the impact of the TPBARs on the radiation sources is discussed in Section 2.12.2.

SRP 12.3 - 12.4
Radiation Protection Design Features: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 20.1(c), 20.101, 20.103, 20.104, 20.203, 20.207; 10 CFR

50.34; 10 CFR 70.24; and 10 CFR 50, Appendix A, GDC 19 and 61. They deal with making reasonable efforts to maintain radiation exposures ALARA; design features, shielding, ventilation, monitoring, and dose assessment to control occupational radiation doses to individuals in restricted areas; design features, ventilation, monitoring, and dose assessment to control intake of radioactive materials in restricted areas; limiting exposure of minors in restricted areas; posting of areas containing radioactivity; securing licensed materials against unauthorized removal; adequate protection to permit access to areas necessary for occupancy after an accident without personnel receiving exposures in excess of 5 rems to the whole body or the equivalent to any part; occupational radiation protection aspects of fuel storage, handling, radioactive waste and other systems to assure adequate safety during normal and postulated accident conditions with suitable shielding, containment and filtering systems; and procedures and criteria for monitoring for criticality accidents involving special nuclear material. An evaluation of the impact of the TPBARs on the dose assessment is discussed in Section 2.12.3.

SRP 12.5
Operational Radiation Protection Program: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 19.12, 10 CFR 20.1(c), 20.101, 20.103, 20.105, 20.201, 20.202, 20.203, 20.204, 20.205, 20.207, 20.401, 20.402, 20.405, 20.408; 10 CFR 50.34; and 10 CFR 50, Appendix A, GDC 64. They deal with providing instructions to workers; making reasonable efforts to maintain radiation exposures ALARA; design features, shielding, ventilation, monitoring, and dose assessment to control occupational radiation doses to individuals in restricted areas; design features, ventilation, monitoring, and dose assessment to control intake of radioactive materials in restricted areas; control of radiation doses to individuals in unrestricted areas; performance of surveys; providing appropriate personnel monitoring equipment to individuals entering restricted areas; posting of areas containing radioactivity; appropriate handling of packages containing certain quantities of radioactive materials; securing licensed materials against unauthorized removal; maintaining records for individuals exposed to radiation in restricted areas; reports of loss or theft of licensed material; requirements for written reports concerning individual exposures in excess of regulatory limits; reports on exposure to terminated individuals; and appropriate monitoring for the reactor containment atmosphere and spaces containing components for recirculation of LOCA fluids. Section 2.12.4 contains a discussion of the expected impact of the TPBARs on this program.

2.12.2 RADIATION SOURCES (SRP 12.2)

2.12.2.1 Acceptance Criteria

The acceptance criteria for the SAR information associated with radiation sources in normal operations, anticipated operational occurrences, and accident conditions are based on meeting the relevant requirements specified in 10 CFR Part 50.34 and if the SAR contains sufficient information related to the relevant requirements of 10 CFR 20 and 10 CFR 50, General Design Criteria 61 - "Fuel storage and handling and radioactivity control". The pertinent sections of 10 CFR 20 are paragraph 20.101 - "Exposure of individuals to radiation in restricted areas" (superseded by Subpart C of the revised 10 CFR 20 - "Occupational Dose Limits") as it relates to limiting radiation doses to protect individuals in restricted areas, paragraph 20.103 - "Exposure

of individuals to concentrations of radioactive material in air, in restricted areas (superseded by Subpart C of the revised 10 CFR 20 -"Occupational Dose Limits") as it relates to limiting average concentrations of radioactive materials, paragraph 20.104 - "Exposure of minors" (superseded by paragraph 20.1207 - "Occupational dose limits for minors") as it relates to limiting radiation exposure of minors, paragraph 20.106 - "Concentrations in effluents to unrestricted areas" (superseded by Subpart D of the revised 10 CFR 20 - "Radiation Dose Limits for Individual Members of the Public") as it relates to radiation levels and radioactive materials concentrations within components of the waste treatment systems, and paragraph 20.207 - "Storage of licensed materials" (superseded by Subpart I of the revised 10 CFR 20 - "Storage and Control of Licensed Material") as it relates to securing licensed materials against unauthorized removal.

2.12.2.2 Evaluation

As described in Section 2.11.2, the impact of operation with a TPC is expected to have a negligible impact on the design basis and realistic fission and corrosion product sources and the treatment of these isotopes in liquid and gaseous waste. Further, the TPC is not expected to impact the fission product source terms that are used for shield design, equipment qualification, systems design and accident dose analyses.

The design basis tritium sources are expected to increase the amount of tritium that is discharged annually by a factor of about five, i.e., from about 890 Ci/year to the TPC design basis value of approximately 4300 Ci/year. This additional tritium inventory will not affect the ability of the plant to meet the applicable regulatory requirements relative to radioactivity in effluents to unrestricted areas (10 CFR Part 20), the "as low as is reasonably achievable" criterion (10 CFR Part 50 - Appendix I), and waste management systems and the control of releases of radioactive materials to the environment (GDC 60). Even if the postulated failure of two TPBARs is considered, which would result in the release of an additional 20,000 Ci of tritium, waste management operations using existing plant equipment can be conducted such that applicable regulatory limits are not exceeded.

2.12.2.3 Conclusion

The impact of the TPC on radiation sources was described in Section 2.11.2.

2.12.3 RADIATION PROTECTION DESIGN FEATURES AND DOSE ASSESSMENT (SRP 12.3-4)

2.12.3.1 Acceptance Criteria

The acceptance criteria for the SAR information associated with radiation protection design features, taking into account design dose rates, anticipated operational occurrences, and accident conditions are based on meeting the relevant requirements specified in 10 CFR Part 50.34 and if the SAR contains sufficient information related to the relevant requirements of 10 CFR 20, 10 CFR 50, Appendix A, and 10 CFR 70. The pertinent sections of 10 CFR 20 are paragraph 20.1(c) - "Purpose" (superseded by the revised 10 CFR 20.1101 - "Radiation Protection Programs") as it relates to radioactivity in effluents to persons involved in

maintaining radiation exposures As Low As Is Reasonably Achievable, paragraphs 20.101 - "Exposure of individuals to radiation in restricted areas" (superseded by Subpart C of the revised 10 CFR 20 - "Occupational Dose Limits") and paragraph 20.103 - "Exposure of individuals to concentrations of radioactive material in air, in restricted areas (superseded by Subpart C of the revised 10 CFR 20 -"Occupational Dose Limits") as they relate to controlling occupational radiation exposures to individuals in restricted areas by design features, shielding, ventilation, monitoring, and dose assessment, paragraph 20.104 - "Exposure of minors" (superseded by paragraph 20.1207 - "Occupational dose limits for minors") as it relates to limiting radiation exposure of minors, paragraph 20.203 - "Cautions, signs, labels, signals, and controls" (superseded by Subpart J of the revised 10 CFR 20 - "Precautionary Procedures") as it relates to posting of radiation areas and other indications necessary to identify and quantify the presence of radioactive materials in an area, and paragraph 20.207 - "Storage of licensed materials" (superseded by Subpart I of the revised 10 CFR 20 - "Storage and Control of Licensed Material") as it relates to securing licensed materials against unauthorized removal. The pertinent sections of 10 CFR 50, Appendix A are GDC 19 - "Control Room" as it relates to adequate radiation protection and access to areas necessary after an accident without exceeding a whole body dose of 5 rems, and GDC 61 - "Fuel storage and handling and radioactivity control" as it relates to occupational radiation protection aspects of fuel storage, handling, radioactive waste, and other systems designed to assure adequate safety during normal and postulated accident conditions. The pertinent section of 10 CFR 70 is Section 70.24 "Criticality Accident Requirements" as it relates to procedures and criteria for monitoring for criticality accidents involving special nuclear materials.

2.12.3.2 Evaluation

The increased inventory of tritium released to the RCS from the TPBARs and the additional IFBA fuel rods, has been evaluated to ensure that normal releases can be maintained within prescribed regulatory limits and that ALARA considerations are addressed. These criteria are related in that if all of the tritium is retained in the plant, rather than released in plant effluents, the tritium concentrations can create undesirable radiological conditions. Important plant activities that can be affected include,

a) containment access during power operation, and

b) refueling operations

To permit containment access during refueling operations and during normal plant operation, the tritium concentration in the containment atmosphere must be limited to prevent excessive personnel exposures.

During power operation, leakage from the primary system into containment could result in high concentrations of tritium and during refueling, the tritium contained in the RCS is dispersed to the refueling cavity, fuel transfer canal, and spent fuel pit. The tritiated water evaporated to the air results in worker radiation dose due to inhalation and absorption through the skin. The derived air concentration (DAC) limit corresponding to the annual limit on intake (ALI) is 2×10^{-5} μCi/cc, for occupational exposures (2000 hours of exposure per year).

For short-term access to the containment during power operation, an upper limit of 3.5 μCi/g in the RCS is generally considered, and is based on the assumption that the RCS is the only source of leakage to the containment. During refueling operations, a refueling water activity concentration of about 2.5 μCi/g is expected to result in containment air tritium concentration which will not exceed the permissible operating personnel dose limits.

A negligible increase in the annual worker radiological exposure due to operation with TPBARs is anticipated, since the recommended procedure is to adjust the plant discharge of the tritiated primary coolant such that buildup of activity concentrations in the plant water volumes do not approach levels that impact worker radiation dose and/or worker efficiency (e.g., levels that would mandate plastic protective clothing, respirators, etc.). The adjustment in plant discharge will depend, in part, on the actual permeation of tritium from the TPBARs, as well as the amount of normal systems leakage at any given point in time and normal waste discharge practices for a particular plant.

The above evaluation of the impact of TPC operation on the reference plant with the design basis release of 1 Ci/rod-yr from the TPBARs indicates that the expected plant discharges would be increased from 10 to approximately 17.5 RCS volumes per year with about a 75% increase in the number of batch releases. This is not expected to have a major impact on liquid waste management and plant operations at the reference plant. Further, this increase in plant liquid discharges maintains primary coolant tritium concentrations at current levels and does not result in significant increases in off-site doses (i.e, generally less than 2% of the limiting dose to a member of the public from liquid effluents and no increase in dose from airborne discharges).

The only potential source of additional exposure associated with TPC operation that has been identified is that associated with worker radiological exposure due to increased fuel and TPBAR handling activities. The TPC core design is based on 140 fresh feed assemblies for the equilibrium cycle. The remaining 53 assemblies are once-burned assemblies with control rods. Thus, the refueling operation would involve the following basic steps:

1. off-load of all (193) assemblies

2. removal of TPBARs from 53 assemblies

3. transfer of control rods to the 53 once-burned assemblies

4. core loading of 140 fresh assemblies and 53 once-burned assemblies with control rods

Most of the handling activities are performed from the bridges above the refueling cavity and spent fuel pit. The source of radiation exposure is from corrosion products in the water that result in radiation fields that are typically in the range of 1-5 millirem/hr at the occupied locations above the water surfaces. It is assumed that the TPBARs arrive already loaded in the new fuel assemblies, or are loaded in the new fuel handling area and that the associated Occupational Radiation Exposure (ORE) is small and unchanged from that with a conventional core.

Current operating plants generally off-load the entire core each refueling. Thus, the off-loading operations and associated ORE are the same, regardless of whether TPBARs are used or not. The difference in the ORE for the remaining handling operations, i.e., with TPBARs versus a normal core refueling, is highly plant dependent. In the case of some plants, all of the assemblies are loaded with either control rods, burnable poison rods, or thimble plugging devices. Since feed assemblies constitute roughly 40-50% of the core in a conventional plant, the total number of handling operations (i.e., control rod transfer, burnable poison removal and insertion, thimble plug shuffling) is expected to exceed that associated with a TPC. However, other plants (including the reference plant) do not use thimble plugging devices. In this case, the differences in handling operation would be the difference between the handling operations associated with 53 assemblies (i.e. the number of TPBARs that need to be removed for control rod installation) and the number of conventional burnable poison rods that must be removed and installed in a conventional core.

In the worst case scenario, there would be no conventional burnable poison rods and limited control rod shuffling, as well as no thimble plugging devices. The total ORE projected for the additional TPBAR handling is estimated based on 2 people for one 12-hour shift in a 2.5 millirem/hour radiation field. ORE associated with control rod handling is estimated based on 2 people for 10 hours in a similar radiation field. The total dose is 110 millirem per fuel cycle which equates to roughly 0.073 person-rem/year (for 18-month fuel cycles). This value is only 0.05% of the average annual ORE reported for the reference plant (approximately 150 man-rem/year based on a 3-year rolling average).

2.12.3.3 Conclusion

A negligible increase in the annual worker radiological exposure due to operation with TPBARs is anticipated, since the recommended procedure is to adjust the plant discharge of the tritiated primary coolant such that buildup of activity concentrations in the plant water volumes do not approach levels that impact worker radiation dose and/or worker efficiency. The only potential source of additional exposure associated with TPC operation that has been identified is that associated with worker radiological exposure due to increased fuel and TPBAR handling activities. The total associated dose is 110 millirem per fuel cycle which equates to roughly 0.073 person-rem/year (for 18-month fuel cycles). This value is only 0.05% of the average annual ORE reported for the reference plant (approximately 150 man-rem/year based on a 3-year rolling average).

2.12.4 OPERATIONAL RADIATION PROTECTION PROGRAM (SRP 12.5) EVALUATION

2.12.4.1 Acceptance Criteria

The acceptance criteria for the SAR information associated with operational aspects of the radiation protection program is based on meeting the relevant requirements specified in 10 CFR Part 50.34 and if the SAR contains sufficient information related to the relevant requirements of 10 CFR 19, 10 CFR 20, and 10 CFR 50 - Criteria 64 - "Monitoring Radioactivity Releases". The pertinent areas subject to review include:

- Organization

- Equipment, Instrumentation, and Facilities
- Procedures

2.12.4.2 Evaluation

The reference plant, as with all operating PWRs, currently has an operating Health Physics organization along with appropriate equipment, instrumentation, facilities and procedures in place for monitoring of tritium both within the plant and in plant releases. Operations with TPC can increase the amount of tritium within the plant and in plant releases. However, changes in the current Health Physics Program are not anticipated.

2.12.4.3 Conclusions

Plant operations with TPCs will not require changes to the existing operational radiation protection program.

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2.13 CONDUCT OF OPERATIONS

2.13.1 INTRODUCTION

The sections in Chapter 13 of the SRP deal with the various aspects of plant operations, including organization, training, emergency planning, operation review, procedures, and security. An evaluation of the impact of incorporating TPBARs on the safeguards and security for the reference plant is presented in Section 2.13.2, below.

SRP 13.1.1
Management and Technical Support Organization: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.40b. They deal with the adequacy of the organization and resources to provide offsite technical support for the operation of the facility. Incorporation of the TPBARs will have no impact on the acceptability of management and technical support resources since these organizations are not required to change.

SRP 13.1.2 - 13.1.3
Operating Organization: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.40b and 10 CFR 50.54j, k, l, and m. They deal with the technical qualification of the licensee and operator requirements. Incorporation of the TPBARs will have no impact on the technical qualification of the plant personnel since personnel changes are not required.

SRP 13.2.1 - 13.2.2
Training: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.54i, j, k, l and m; 10 CFR 55.21, 55.22, 55.23; 10 CFR 55, Appendix A; Regulatory Guide 1.8; NUREG-0094; 10 CFR 19.12; 10 CFR 50.34a and b; and 10 CFR 50.40b. They deal with specific requirements for plant staff training. Each plant has a training program in place to meet these requirements, and any plant incorporating TPBARs will need to modify their program to assure its continued adequacy, and to include additional training required for handling and/or monitoring the TPBARs.

SRP 13.3
Emergency Planning: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.47b as elaborated in 10 CFR 50, Appendix E(IV); NUREG-0654, Revision 1; and NUREG-0696. They deal with the development of plans for responding to emergencies. Each plant has a comprehensive emergency plan in place to meet these requirements, and any plant incorporating TPBARs should review and adjust that plan as appropriate.

SRP 13.4
Operation Review: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.40b. They deal with the technical qualification of the licensee as demonstrated in the program defined for review of plant operations and changes. The incorporation of the TPBARs will have no impact, since this review program will continue.

SRP 13.5.1 - 13.5.2
Administrative, Operating and Maintenance Procedures: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.40b and 10 CFR 50.34. They deal with the technical qualification of the licensee as demonstrated in the program defined for developing, maintaining and changing procedures. The incorporation of the TPBARs will have no impact on this program for dealing with procedures, although specific plant procedures (including operating procedures, maintenance procedures, and Emergency Operating Procedures) under this program will need to be reviewed and possibly revised for incorporation of the TPBARs. This will be dealt with by the candidate plant.

SRP 13.6
Physical Security: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.34c; 10 CFR 73.55; 10 CFR 73, Appendices B and C; 10 CFR 25; 10 CFR 75; and 10 CFR 95. They deal with the physical security organizations; physical barriers to protect vital equipment; control of access to protected areas; intrusion alarms; communications between security personnel; testing and maintenance of alarms, communication equipment, access control equipment, physical barriers, and other security-related equipment; response requirements; selection and training of security personnel; and safeguards contingency plans. An evaluation of the impact of the TPBARs is presented in Section 2.13.2.

2.13.2 SAFEGUARDS AND SECURITY (SRP 13.6) EVALUATION

2.13.2.1 Introduction

To protect national security interests and DOE's investment, the host utility will account for each tritium-producing burnable absorber rod (TPBAR) and TPBAR assembly from receipt until off-site shipment. The TPBARs and documentation necessary to support the preparation of the host utility safety evaluation will be classified Confidential Restricted Data (CRD). Classified material and documents require safeguards measures to prevent diversion and or unauthorized access to or disclosure of classified information.

2.13.2.2 Materials Control and Accountability

DOE requires that the TPBARs be controlled and accounted for due to the initial presence of lithium-6 (^{6}Li) and, post-irradiation, tritium (^{3}H). The host utility will provide the irradiation history to DOE to allow analysis of ^{6}Li conversion and ^{3}H production for accounting purposes; hence, the utility need account for only the physical assemblies. DOE material control and accountability requirements for the control of ^{6}Li and ^{3}H in the TPBARs are satisfied by meeting the NRC requirements for fuel.

To accomplish the physical control and accountability, the TPBARs will be subjected to the same materials control and accountability as nuclear fuel. Each TPBAR will have a unique number engraved or etched on the top end plug. The TPBAR assemblies will be identified by a unique serial number on the hold-down assembly such as is currently used on fuel inserts. These serial numbers will be used in shipping documentation and irradiation records so that they can be tracked and accounted for. The host utility's internal control and accountability

procedures for fuel will be adequate and consistent with the material control and accountability for the TPBARs and assemblies.

The materials and accountability of the TPBARs and assemblies for initial receipt and off-site preparation will be conducted in a manner consistent with the host utility's administrative guidelines for handling of new and spent fuel inserts. Carriers who meet Department of Transportation requirements for shipment of nuclear material will be used for the transportation of the TPBARs to and from the site.

2.13.2.3 Physical Security of Classified Hardware

The TPBARs require physical protection commensurate with their classification as CRD. Under the requirements for a reactor operating license, the selected utility will have performed a vulnerability assessment meeting the requirements of 10 CFR 73 for physical protection of vital equipment. DOE will need to assess the selected utility's vulnerability assessment against DOE's design basis threat criteria. The design basis threats of the DOE for CRD and the NRC for fuel are similar; consequently, little or no modification of the utility's physical security is expected. Any modification to physical security deemed necessary to meet DOE's established criteria will be in accordance with the Memorandum of Understanding between DOE and the NRC (Reference 1).

Prior to irradiation, the TPBARs will require an escort by DOE-cleared personnel during movement of the assemblies. These include movement from the carrier to the new fuel storage racks, or the spent fuel pool, and from there to the reactor. As the TPBARs are visually unclassified, there is no need to protect the TPBARs or TPBAR assemblies from casual observation or visual inspection by personnel who have not been granted access to classified information.

While the TPBAR assemblies are stored in the new fuel storage racks or in the spent fuel pool, a suitable level of physical protection will be provided. The vulnerability assessment performed on the selected utility will determine whether existing operations meet DOE's protection requirements. Comparison of existing procedures indicates that little or no modification to protection procedures will be needed at the host utility. While the TPBAR assemblies are in the reactor with the reactor head bolted, they will be considered secure and no escort by DOE-cleared personnel will be required.

2.13.2.4 Control of Classified Documents and Hardware

Under the requirements for a reactor operating license, the site must have a physical security plan meeting the requirements of 10 CFR Part 73 for physical protection of vital equipment and control of safeguards information. These requirements include a physical security organization, physical barriers, access controls, detection aids, communication, procedures for testing and maintenance of security equipment, and safeguards contingency plan. Based on the vulnerability assessment performed on the selected utility, the site will have to develop and implement procedures (under DOE assistance and guidance) for handling classified hardware.

Personnel will be granted access to classified information and hardware by DOE under 5 CFR 732 and 10 CFR 710 to meet Executive Orders 12958 (Reference 2) and 12968 (Reference 3). These controls will meet or exceed those that would be required to meet 10 CFR Part 25 and 10 CFR Part 95 and shall be deemed to satisfy 10 CFR 50.37 requirements for licensee access to restricted data.

In accordance with the Memorandum of Understanding between DOE and the NRC (Reference 1), DOE has reached an agreement with the NRC that NRC licensees seeking access authorization from the DOE, based on their participation in the CLWR Tritium Project, do not require additional access authorization (Reference 4). The NRC has acknowledged this agreement and confirmed that conformance with DOE requirements relative to access authorization satisfies corresponding NRC requirements contained in 10 CFR Parts 25 and 95 as well as the facility operating license provisions contained in 10 CFR 50.37 relative to licensee access to restricted data (Reference 5). The DOE will perform required background investigations appropriate to the level of access authorization being sought for the personnel, and no additional NRC action is required for personnel access authorizations relative to the CLWR Project.

Security facility approval will be coordinated between DOE and NRC to meet applicable regulatory requirements. DOE will perform facility reviews to ensure that DOE classified hardware (for example, TPBAR assemblies) to be handled or stored at the facility is appropriately safeguarded. The granting of access authorization and the coordination of facility approval with NRC for a DOE program in an NRC licer ed facility is consistent with the direction provided in Executive Order 12968 and the NRC roposed Rule on Access to and Protection of Classified Information" (61FR40555).

The area used for storage of any classified documents at the host utility will require that the designated area be an approved facility per 10 CFR 1016 and excluded from IAEA inspections. As the TPBARs are not visually classified, this exclusion does not apply to the hardware.

2.13.2.5 References

1. Memorandum of Understanding between Department of Energy and the Nuclear Regulatory Commission under the Provisions of The National Industrial Security Program, dated September 19, 1996.
2. Executive Order 12958, "Classified National Security Information," dated April 17, 1995.
3. Executive Order 12968, " Access to Classified Information," dated August 4, 1995.
4. Letter, S.M. Sohinki to J.H. Wilson, October 4, 1996, "DOE Clearances for NRC Licensees Supporting Tritium Program; Project No. 697."
5. Letter, T.T. Martin to S.M. Sohinki, November 1, 1996, "DOE Clearances for NRC Licensees Supporting Tritium Program."

2.14 INITIAL TEST PROGRAM

2.14.1 INTRODUCTION

There is only one review plan in this chapter of the SRP. The acceptance criteria in SRP 14.2 (Initial Plant Test Program - Final Safety Analysis Report) are based on the relevant requirements of 10 CFR 30.53; 10 CFR 50.34(b)(6)(iii); 10 CFR 50, Appendix B, Section XI; and 10 CFR 50, Appendix J, Section III.A.4, and are supplemented by Regulatory Guides 1.18, 1.20, 1.30, 1.37, 1.41, 1.52, 1.68, 1.72, 1.79, 1.80, 1.95, 1.108, 1.116, 1.128, 1.139 and 1.140. They deal with the definition of a test program to demonstrate that components and systems operate in accordance with design requirements. Section 2.14.2 provides a discussion of the impact of TPBARs on the initial test program.

2.14.2 INITIAL TEST PROGRAM (SRP 14.2)

The initial plant startup test program was evaluated for the condition of loading a full core complement of fresh fuel assemblies and TPBARs in each core location that does not contain a rod cluster control assembly. There are no modifications to the reactor coolant system or its support systems for handling and processing waste effluents. Operation of the plant with the TPC is not significantly different from the operation of a non-tritium producing core.

There are eight items related to the test program which are specifically identified in SRP 14.2. The impact of the incorporation of TPBARs is discussed below for each item.

1. **Summary of Test Programs and Objectives**

   The majority of testing required by Regulatory Guide 1.68 is not impacted by the TPC. The tests that are potentially impacted are the core physics tests performed during the time between core loading and ascent to full power. The applicant should update the summary for the core loading process, as it will be different for the tritium producing core than it was for the Cycle 1 core loading.

   The tests performed for initial criticality and power ascension are greatly abbreviated for reload cores. Industry guidance is provided in ANSI/ANS 19.6.1, "Reload Startup Physics Tests For Pressurized Water Reactors", for verifying the nuclear characteristics of a commercial pressurized water reactor core. The tests described in this ANSI standard appear to be sufficient for verification of the nuclear characteristics of the tritium producing core. Each applicant should review, and if necessary, update the summaries for the tests performed during initial criticality and power ascension tests, in accordance with ANSI/ANS 19.6.1.

2. **Test Procedures**

   The test procedures for loading a full core complement of fresh fuel assemblies and TPBARs should be developed with special attention given to the expected source range count during and following core loading. The test procedures for reload startup physics tests should be acceptable in their current form.

3. **Test Program's Conformance with Regulations and Regulatory Guides**

   The initial plant startup program detailed the conformance with regulations and regulatory guides including where exceptions were taken to regulations and regulatory guides. The incorporation of TPBARs in the core should not constitute deviation from the initial program for their performance as a core power shape control insert. However, the applicant should be prepared to provide details on core loading and monitoring as the loading of a full core complement of fresh fuel assemblies resembles the initial cycle core loading.

4. **Utilization of Reactor Operating and Testing Experiences in the Development of the Test Program**

   The operating experience obtained from the CLWR TPBAR Lead Test Assembly program in Watts Bar Cycle 2 should be considered by the applicant in the review of the core loading and reload startup testing procedures.

5. **Trial Use of Plant Operating and Emergency Procedures**

   There are no changes expected to the operating, surveillance or emergency procedures that would require integration into the startup testing program. Additional procedures may be required at the facility for security and safeguards (see Section 2.13.2). Furthermore, some input assumptions for LOCA related releases of RCS water outside of containment may need to be reduced to keep the actions described in the procedure valid (on-site dose related).

6. **Initial Fuel Loading and Initial Criticality**

   Refer to the comments provided for items 1, 2 and 3.

7. **Test Program Schedule and Sequence**

   The schedule requirements given in this section are not relevant for the testing program required for the TPC.

8. **Individual Test Descriptions/Abstracts**

   For the tests described in items 1, 2 and 3, the relationship between the test and the TPBARs should be clearly defined. An abstract for the core loading procedure should indicate that the production of tritium is increased by loading the TPBARs in fresh fuel assemblies, therefore the core loading and monitoring will be different from the initial cycle and previous reload cycles. The physics tests performed are direct measurements of the performance of the core with the TPBARs installed and the abstracts should describe them as such.

Although not explicitly part of SRP 14.2, another item to consider is the change in reactivity holdup in the TPBARs that occurs following irradiation and extended shutdown. As described in Section 2.4.3.8, an extended midcycle outage will impact the operation of the plant upon restart.

2.15 ACCIDENT ANALYSIS

2.15.1 INTRODUCTION

The sections in Chapter 15 of the SRP are related to the analyses of a specific set of anticipated operational occurrences and postulated accidents. These analyses include not only the transient analyses, but also the radiological consequences of those accidents which could result in the release of radioactive materials. Table 2.15.1-1 lists the applicable transients for the reference plant. Each transient and each potential radiological release was either evaluated or reanlyzed to determine the impact of the TPBARs. Section 2.15.2 provides the evaluations and analyses of the non-LOCA transients with TPBARs listed in Table 2.15.1-1, except for the anticipated transients without scram (ATWS) which are discussed in Section 2.15.7. The impact of TPBARs on the inadvertant loading event is discussed in Section 2.15.3. Section 2.15.4 provides an evaluation of the impact of TPBARs on the steam generator tube rupture event, while the impact on small and large break loss of coolant accidents (LOCAs) is described in Section 2.15.5. The impact of TPBARs on the radiological consequences of accidents is discussed in Section 2.15.6.

SRP 15.1.1 - 15.1.4
Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 10, 15 and 26, and TMI Action Plan items II.E.5.1 and II.E.5.2 of NUREG-0718. They deal with the design of the reactor coolant system with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences; the design of the reactor coolant system and its associated auxiliaries with appropriate margin to assure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences; and the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded. An evaluation of the impact of TPBARs on the Feedwater Temperature Decrease, Steam Flow Increase and Inadvertent Opening of a Steam Generator Relief or Safety Valve is discussed in Section 2.15.2.5. An analysis of the impact of TPBARs on the Feedwater Flow Increase was performed and is described in Section 2.15.2.5.1.

SRP 15.1.5
Steam System Piping Failures Inside and Outside of Containment: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 27, 28, 31 and 35 and NUREGs 0694, 0718, and 0737. They deal with the design of the reactor coolant system with appropriate margin to assure that specified acceptable fuel design limits are not exceeded and that control rod insertability and the capability to cool the core are maintained; the design of the reactor coolant system with sufficient margin to assure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized; the design of the reactor cooling system and associated auxiliaries to provide abundant emergency core cooling; and to maintain adequate decay heat removal and reactor coolant pump integrity and operation. An evaluation of the impact of TPBARs on the Steam System Piping Failure is presented in Section 2.15.2.5.

SRP 15.1.5, Appendix A
Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100. They deal with the exposure guidelines for whole-body and thyroid doses at the exclusion area and the low population zone outer boundaries for a postulated Main Steam Line Break (MSLB) outside containment. The impact of the TPBARs on the radiological consequences for the Main Steam Line Break is discussed in Section 2.15.6.4.

SRP 15.2.1 - 15.2.5
Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve, and Steam Pressure Regulator Failure (Closed): The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 10, 15 and 26. They deal with the design of the reactor coolant system with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences; the design of the reactor coolant system and its associated auxiliaries with appropriate margin to assure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences; and the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded. An evaluation of the impact of TPBARs on these events is presented in Section 2.15.2.6.

SRP 15.2.6
Loss of Nonemergency AC Power to the Station Auxiliaries: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 10, 15 and 26, and TMI Action Plan Items II.E.1.1, II.E.1.2, and II.K.2(1) of NUREGs-0718 and -0737. They deal with the design of the reactor coolant system with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences; the design of the reactor coolant system and its associated auxiliaries with appropriate margin to assure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences; the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded; and the performance requirements of the auxiliary feedwater system for the loss of nonemergency ac power event. An evaluation of the impact of TPBARs on this event is presented in Section 2.15.2.6.

SRP 15.2.7
Loss of Normal Feedwater Flow: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 10, 15 and 26, and TMI Action Plan Items II.E.1.1, II.E.1.2, and II.K.2(1) of NUREGs-0718 and -0737. They deal with the design of the reactor coolant system with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences; the design of the reactor coolant system and its associated auxiliaries with appropriate margin to assure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences; the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded; and the performance requirements of the auxiliary feedwater system for

the loss of normal feedwater flow event. An evaluation of the impact of TPBARs on this event is presented in Section 2.15.2.6.

SRP 15.2.8
Feedwater System Pipe Breaks Inside and Outside of Containment: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 27, 28, 31 and 35; 10 CFR 100; and TMI Action Items II.E.1, II.K.2.1, II.E.1.2, II.K.3.5, II.K.2.16 and II.K.3.25 of NUREGs-0718 and -0737. They deal with the design of the reactor coolant system with appropriate margin to assure that specified acceptable fuel design limits are not exceeded and the capability to cool the core is maintained; the design of the reactor coolant system with sufficient margin to assure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized; the design of the reactor cooling system and associated auxiliaries to provide abundant emergency core cooling; the calculation of doses at the site boundary; and the capability to maintain adequate decay heat removal and reactor coolant pump integrity and operation. An evaluation of the impact of TPBARs on this event is presented in Section 2.15.2.6.

SRP 15.3.1 - 15.3.2
Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 10, 15 and 26. They deal with the design of the reactor coolant system with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences; the design of the reactor coolant system and its associated auxiliaries with appropriate margin to assure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences; and the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded. An evaluation of the Partial Loss of Forced Reactor Coolant Flow is presented in Section 2.15.2.7.1, and an analysis of the Complete Loss of Forced Reactor Coolant Flow is described in Section 2.15.2.7.2.

SRP 15.3.3 - 15.3.4
Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 27, 28 and 31, and 10 CFR 100. They deal with the design of the reactor coolant system with appropriate margin to assure that the capability to cool the core is maintained; the design of the reactor coolant system with sufficient margin to assure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized; and the calculation of doses at the site boundary. The results of the locked rotor transient analysis with TPBARs are described in Section 2.15.2.7.3, while the shaft break is discussed in Section 2.15.2.7.4. The radiological consequences are discussed in Section 2.15.6.

SRP 15.4.1
Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 10, 20 and 25. They deal with the requirement that specified

acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences; the automatic initiation by the protection system of appropriate systems to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences; and the design of the reactor protection system to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system. The results of the analysis of an Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Condition with TPBARs are described in Section 2.15.2.8.1.

SRP 15.4.2
Uncontrolled Control Rod Assembly Withdrawal at Power: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 10, 20 and 25. They deal with the requirement that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences; the automatic initiation by the protection system of appropriate systems to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences; and the design of the reactor protection system to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system. An evaluation of the impact of TPBARs on this event is discussed in Section 2.15.2.8.

SRP 15.4.3
Control Rod Misoperation (System Malfunction or Operator Error): The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 10, 20 and 25. They deal with the requirement that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences; the automatic initiation by the protection system of appropriate systems to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences; and the design of the reactor protection system to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system. An evaluation of the impact of TPBARs on this event is discussed in Section 2.15.2.8.

SRP 15.4.4
Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 10, 15, 20, 26 and 28. They deal with the design of the reactor coolant system with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences; the design of the reactor coolant system and its associated auxiliaries with appropriate margin to assure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences; and the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded. An evaluation of the impact of TPBARs on this event is discussed in Section 2.15.2.8.2.

SRP 15.4.6
Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 10, 15 and 26. They deal with the design of the reactor coolant system with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences; the design of the reactor coolant system and its associated auxiliaries with appropriate margin to assure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences; and the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded. An evaluation of the impact of TPBARs on this event is discussed in Section 2.15.2.8.

SRP 15.4.7
Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 13, and 10 CFR 100. They deal with the instrumentation and controls provided to monitor variables over anticipated ranges for normal operations, anticipated operational occurrences, and for accident conditions, and with the offsite consequences resulting from reactor operations with an undetected misloaded fuel assembly. Specific requirements are 1) the plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel loading errors after fueling operations, and 2) in the event the error is not detectable by the instrumentation system and fuel rod failure limits could be exceeded during normal operation, the offsite consequences should be a small fraction of the 10 CFR 100 guidelines. The impact of TPBARs on various misloading scenarios is described in Section 2.15.3.

SRP 15.4.8
Spectrum of Rod Ejection Accidents: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 28. They deal with the requirement that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding nor cause sufficient damage to impair significantly the capacity to cool the core. The results of the Rod Ejection transient analysis with TPBARs are presented in Section 2.15.2.8.3.

SRP 15.4.8, Appendix A
Radiological Consequences of a Control Rod Ejection Accident: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100. They require that the calculated whole-body and thyroid doses at the exclusion area and the low population zone boundaries be well within (defined as 25% of) the exposure guidelines in 10 CFR 100, paragraph 11. The impact of the TPBARs on the radiological consequences of a control rod ejection accident is discussed in Section 2.15.6.4.

SRP 15.5.1 - 15.5.2
Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 10, 15 and 26. They deal with the

design of the reactor coolant system with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences; the design of the reactor coolant system and its associated auxiliaries with appropriate margin to assure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences; and the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded. An evaluation of the impact of TPBARs on these events is presented in Section 2.15.2.9.

SRP 15.6.1
Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 10, 15 and 26, and TMI Action Plan Items II.K.3.1, II.K.3.5, and II.K.3.25 of NUREGs-0718 and -0737. They deal with the design of the reactor coolant system with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences; the design of the reactor coolant system and its associated auxiliaries with appropriate margin to assure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences; the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded; and the capability to maintain adequate decay heat removal and reactor coolant pump integrity and operation. An evaluation of the impact of TPBARs on this event is described in Section 2.15.2.10.

SRP 15.6.2
Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 55 and 10 CFR 100.11. They deal with the isolation provisions for small diameter lines connected to the primary system which penetrate containment, and the allowable radiological consequences of a break in such a line. Specifically, the calculated whole-body and thyroid doses at the exclusion area and the low population zone boundaries must be a small fraction (defined as 10%) of the exposure guidelines in 10 CFR 100.11. The impact of the TPBARs on the radiological consequences of a small line failure outside containment is discussed in Section 2.15.6.4.

SRP 15.6.3
Radiological Consequences of Steam Generator Tube Failure: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 100.11. They deal with the allowable radiological consequences of a postulated steam generator tube failure accident. Specifically, the calculated whole-body and thyroid doses at the exclusion area and the low population zone boundaries must not exceed the guideline values for the postulated accident with an assumed preaccident iodine spike in the reactor coolant and for the postulated accident with the highest worth control rod stuck out of the core; and the calculated doses must not exceed a small fraction (defined as 10%) of the exposure guidelines for the postulated accident with the equilibrium iodine concentration for continued full power operation in combination with an assumed accident initiated iodine spike. The evaluation to assess the impact of the TPBARs on the Steam Generator Tube

Failure is described in Section 2.15.4, and the impact on radiological consequences is discussed in Section 2.15.6.4.

SRP 15.6.5 and Appendices A and B
Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50.46; 10 CFR 50, Appendix K; 10 CFR 50, Appendix A, GDC 35; 10 CFR 100; and TMI Action Plan Items II.E.2.3, II.K.3.5, II.K.3.25, II.K.3.30, and II.K.3.31 of NUREGs-0718 and -0737. They deal with the capability of the ECCS equipment to refill the vessel in a timely manner for a LOCA; the provision of redundant ECCS components to adequately cool the core during a LOCA; and the allowable radiological consequences of the most severe LOCA. An evaluation was performed to assess the impact of the incorporation of TPBARs on the design basis LOCAs for the reference plant. This evaluation is described in Section 2.15.5 below, and the radiological consequences are discussed in Section 2.15.6.2.

SRP 15.7.3
Postulated Radioactive Releases Due to Liquid-Containing Tank Failures: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 60, and 10 CFR 20. They deal with the capability of the radioactive waste management systems to control releases of radioactive materials to the environment, and the restrictions on radioactivity in effluents to unrestricted areas. Specifically, tanks and associated components containing radioactive liquids outside containment are acceptable if failure does not result in radionuclide concentrations in excess of the limits in 10 CFR 20, Appendix B, Table II, Column 2, at the nearest potable water supply in an unrestricted area, or if special design features are provided to mitigate the effects of postulated failures for systems not meeting these limits. The impact of TPBARs on the radioactive releases due to tank failures is discussed in Section 2.15.6.4.

SRP 15.7.4
Radiological Consequences of Fuel Handling Accidents: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 61, and 10 CFR 100. They deal with the design of the fuel storage and handling systems to have appropriate containment, confinement and filtering systems, and with the restrictions on radiological consequences of a fuel handling accident. The incorporation of TPBARs does not require changes to the containment, confinement and filtering systems. The impact of the TPBARs on the radiological consequences of these accidents is discussed in Section 2.15.6.3.

SRP 15.7.5
Spent Fuel Cask Drop Accidents: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 61, and 10 CFR 100. They deal with the design of the fuel storage and handling systems to have appropriate containment, confinement and filtering systems, and with the restrictions on radiological consequences of a spent fuel cask drop accident. The incorporation of TPBARs does not require changes to the containment, confinement and filtering systems. At the reference plant, cask handling over the spent fuel pool or the new fuel pit is prevented by interlocks. In addition, because

of the use of a single-failure-proof crane, cask drop is not considered by the reference plant to be a credible accident. Therefore, there are no radiological consequences. For applicant plants which do have this in their design basis, it is not anticipated that the radiological consequences will increase with the incorporation of TPBARs. This is due to the reduction in the iodine source term, as discussed for the fuel handling accident in Section 2.15.6.3. This should be evaluated on a plant specific basis.

SRP 15.8
Anticipated Transients Without Scram (ATWS): The acceptance criteria in SRP 15.8 (Revision 1, July 1981) are based on the relevant requirements of 10 CFR 50, Appendix A, GDC 10, 15, 26, 27 and 29. They deal with the design of the reactor coolant system with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated transients; the design of the reactor protection system with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary will not be exceeded during normal operations including anticipated transients; the requirement for two independent reactivity control systems; the combined capability of the reactivity control systems such that reactivity changes can be reliably controlled to assure that under postulated accident conditions and with the appropriate margin for stuck rods, the core can be cooled; the design of the protection and reactivity control systems to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences. Section 15.8 of the reference plant Final Safety Analysis Report (FSAR) contains additional material related to the regulatory treatment of the ATWS event. For Westinghouse plants, a series of generic studies (References 1 and 2) on ATWS has shown that acceptable consequences will result provided that the turbine is tripped and auxiliary feedwater flow is initiated in a timely manner. The effects of ATWS events are not considered as part of the design basis for transients analyzed in Chapter 15 of the reference plant FSAR. Consistent with the results of the generic studies, the final NRC ATWS rule (Reference 3) requires that Westinghouse-designed plants install ATWS mitigation system actuation circuitry (AMSAC) to initiate a turbine trip and actuate auxiliary feedwater flow, independent of the reactor protection system. The reference plant AMSAC design is described in Section 7.7 of the reference plant FSAR. The impact of the TPBARs on the reference plant response to ATWS is described in Section 2.15.7.

Table 2.15.1-1
Applicable Transients For TPBAR Core Reload Safety Evaluation
(From Reference Plant FSAR)

FSAR SECTION	EVENT	
15.1	INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM	
	15.1.1	Feedwater System Malfunction Resulting in a Decrease in FW Temp.
	15.1.2	Feedwater System Malfunction Resulting in an Increase in FW Flow
	15.1.3	Excessive Increase in Secondary Steam Flow
	15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve
	15.1.5	Steam System Piping Failure
15.2	DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM	
	15.2.2	Loss of External Electrical Load
	15.2.3	Turbine Trip
	15.2.4	Inadvertent Closure of Main Steam Isolation Valves
	15.2.5	Loss of Condenser Vacuum & Other Events Resulting in Turbine Trip
	15.2.6	Loss of Non-Emergency AC Power to the Plant Auxiliaries
	15.2.7	Loss of Normal Feedwater Flow
	15.2.8	Feedwater System Pipe Break
15.3	DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE	
	15.3.1	Partial Loss of Forced Reactor Coolant Flow
	15.3.2	Complete Loss of Forced Reactor Coolant Flow
	15.3.3	Reactor Coolant Pump Shaft Seizure (Locked Rotor)
	15.3.4	Reactor Coolant Pump Shaft Break
15.4	REACTIVITY AND POWER DISTRIBUTION ANOMALIES	
	15.4.1	Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition
	15.4.2	Uncontrolled RCCA Bank Withdrawal at Power
	15.4.3	RCCA Misalignment (Including Drop)
	15.4.4	Startup of an Inactive Reactor Coolant Pump at an Incorrect Temp.
	15.4.6	Chemical and Volume Control System Malfunctions that Result in a Decrease in the Boron Concentration in the Reactor Coolant (Boron Dilution)
	15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position
	15.4.8	Spectrum of RCCA Ejection Accidents
	15.4.9	Steamline Break With Coincident RCCA Withdrawal at Power

Table 2.15.1-1 (continued)
Applicable Transients For TPBAR Core Reload Safety Evaluation
(From Reference Plant FSAR)

FSAR SECTION	EVENT	
15.5	INCREASE IN REACTOR COOLANT INVENTORY	
	15.5.1	Inadvertent Operation of ECCS During Power Operation
	15.5.2	Chemical & Volume Control System Malfunction that Increases Reactor Coolant Inventory
15.6	DECREASE IN REACTOR COOLANT INVENTORY	
	15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve
	15.6.2	Break in Instrument Line or Other Lines from RCS Pressure Boundary that Penetrate Containment
	15.6.3	Steam Generator Tube Failure
	15.6.5	Loss of Coolant Accident
15.8	ANTICIPATED TRANSIENTS WITHOUT TRIP	

2.15.2 SAFETY EVALUATION FOR THE NON-LOCA ACCIDENTS

2.15.2.1 Introduction

This section documents the reload safety evaluation to assess the impact of the TPBARs on the FSAR Chapter 15 Non-LOCA accident analyses for the reference plant. The non-LOCA accidents analyzed for the reference plant FSAR generally follow the format defined by the U.S. NRC in Regulatory Guide 1.70 Revision 3 (Reference 4). A list of the all the accidents (not just non-LOCA) analyzed in Chapter 15 of the reference plant FSAR is shown in Table 2.15.1-1. The general approach taken in the current safety evaluation will be to identify any effects of the TPBARs on the input parameters and methods used to perform the FSAR analyses, and to present event-specific evaluations or analyses that address the impact of the TPBARs on the affected events. If an accident is not affected, this will be noted. The evaluation will be performed using the standard Westinghouse reload evaluation methodology (Reference 5), as discussed below. The performance of the TPBARs in response to the non-LOCA transients is discussed in Section 3.4.

The reference plant FSAR considers four categories of plant conditions, as defined by the American Nuclear Society (ANS) in ANS N18.2-1973. These categories are based on the expected frequency of occurrence for each of the subject events. The basic principle applied in relating design requirements to each of the operating conditions is that the most probable occurrences should yield the least radiological risk to the public, while those extreme situations having the potential for the greatest risk to the public should be those least likely to occur. The event classifications and the associated general acceptance criteria are as follows:

<u>ANS Condition I - Normal Operation:</u> These are occurrences that are expected to occur frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Since Condition I events occur frequently, they must be considered from the point of view of their effect on the consequences of faulted conditions (Condition II, III, and IV). In this regard, analysis of each faulted condition described in the FSAR is generally based on a conservative set of initial conditions corresponding to adverse conditions which can occur during Condition I operation.

<u>ANS Condition II - Faults of Moderate Frequency:</u> These events include most of the faults evaluated in Chapter 15 of the reference plant FSAR. There must be no consequential loss of function of any barrier to the escape of radioactive fission products (no fuel rod failures, reactor coolant system failure, or secondary system overpressurization) as a result of a Condition II fault. Any release of radioactive materials in effluent to unrestricted areas shall be in conformance with 10 CFR 20. By definition, Condition II faults do not propagate to cause a more serious fault, i.e., Condition III or IV events.

<u>ANS Condition III - Infrequent Faults:</u> Condition III events are faults which may occur infrequently during the life of the plant and may result in the failure of only a small fraction of the fuel rods. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion area boundary in accordance with the guidelines of

10 CFR 100. There must be no consequential loss of function of the reactor coolant system or reactor containment barriers.

ANS Condition IV - Limiting (Design Basis) Faults: These faults are not expected to occur but are postulated because their consequence would include the potential for release of significant amounts of radioactive material. Condition IV events are the most drastic faults which must be designed against and they represent the limiting design cases. Following a Condition IV fault, the reactor must be capable of being brought to a safe state, with acceptable heat transfer geometry. There must be no consequential loss of function of systems needed to cope with the event. Condition IV faults are not to cause a fission product release to the environment resulting in doses in excess of the guideline values of 10 CRF 100.

2.15.2.2 Safety Evaluation Methodology

The purpose of the reload safety evaluation is to assess the validity of the existing licensing basis safety analysis for the reference plant. Consistent with the Westinghouse methodology defined in Reference 5, the existing reference plant non-LOCA safety analysis has generally been performed using reload related input parameters selected to bound the expected values for all subsequent cycles. For a given accident, if all safety related parameters, for the specific reload cycle being considered, are bounded by those assumed in the existing analysis, then that analysis continues to define a valid licensing basis for the plant. On the other hand, when any safety related reload parameter is not bounded, further evaluation is necessary. The purpose of this evaluation is to confirm that the margin of safety, as defined in the basis for any technical specification, is not reduced. This bounding analysis concept is a key concept in the NRC-approved Westinghouse reload safety analysis methodology (Reference 5).

In the performance of a reload safety evaluation, the current licensing basis analysis for each accident is examined and the bounding values of the key safety parameters which could be affected by the reload are determined. These bounding parameters form the basis for determining whether the existing safety analysis will continue to bound operation during subsequent cycles. For each reload, including the current reload with the use of TPBARs, values of these key safety parameters are determined for the reload core during the nuclear, thermal and hydraulic and fuel rod design process. Each of these parameters is compared with the existing analysis value to determine if any parameters are out of bounds. If all the parameters are within bounds, the existing analysis remains valid and no new analysis is needed. Should one or more of the key safety parameters be out of bounds, a re-evaluation or re-analysis of any affected accident is performed.

The above discussion has so far considered only those changes to key safety parameters as a result of the reconfiguration of the fuel assemblies in the reload cycle. Accident analysis may also be necessary if there will be any changes to the reactor plant systems or the control or protection systems, whether a direct result of the reload or as a result of the planned outage. All of these changes are considered in the reload safety evaluation for the plant.

The basic methodology and computer codes utilized for the analysis presented here are the same as documented in the reference plant Final Safety Analysis Report (FSAR, Reference 6).

Repetition of the explanatory discussion found in the FSAR will therefore be kept to a minimum.

2.15.2.3 Effect of TPBARs on Analysis Inputs

The non-LOCA safety analysis parameters have been determined for the reference plant reload core design using TPBARs. These parameters were compared to the parameters used in the current applicable safety analysis for the reference plant. This evaluation shows:

1. No changes have been identified in the nominal plant operating conditions (power, coolant temperature, press and flow rate) assumed in the plant safety analysis in order to accommodate the Rs. Therefore, the existing safety analysis cal for the reference plant ar ffected by any changes in plant parameters o the TPBARs.

2. No changes to the reactor core thermal hydraulic characteristics or p eaking factors which could affect the core thermal limits (DNBR and overpower), have n identified result of the use of TPBARs. Therefore, the plant thermal limit protection system setpoints do not change as a result of the TPBARs.

3. The nuclear design and fuel rod design calculations performed for the TPBAR reload core design have identified the following safety analysis parameters as being outside of the bounds of the current applicable reload safety analysis parameters:

   a. The least-negative Doppler-only power defect at the beginning of the cycle has been reduced from 998 pcm to 920 pcm (1 pcm = $1 \times 10^{-5} \Delta k$).

   b. Although not due to the presence of TPBARs, the maximum fuel pellet average and surface temperatures have been increased slightly (maximum of 21°F). These fuel pellet temperature increases reflect the use of a thinner IFBA coating than in the reference plant core design. The reduced th ss IFBA coating is more benign with respect to fuel rod internal pressure, but th y larger pellet-to-cladding gap does increase the predicted fuel temperature.

Due to the reduction in the BOL Doppler power defect, the accidents sensitive to this parameter must be re-evaluated or re-analyzed. The accidents which use this parameter in the FSAR analysis are the uncontrolled RCCA bank withdrawal from subcritical and BOL RCCA ejection (see Table 2.15.1-1). The analysis for these events was repeated and is presented below. Also, in the existing analysis for the reference plant, addressing the hot zero power feedwater malfunction (excessive feedwater flow) event includes the use of results from the uncontrolled RCCA bank withdrawal from subcritical analysis. Therefore, the hot zero power case for the excessive feedwater accident was also reanalyzed.

The increase in maximum fuel temperature has the greatest potential to affect those non-LOCA events for which the assumption of a maximum fuel temperature is conservative and that directly model fuel rod response, using the FACTRAN computer code (see Section 2.15.2.4). These accidents are:

- Partial loss of forced reactor coolant flow
- Complete loss of forced reactor coolant flow
- Startup of an inactive reactor coolant pump at an incorrect temperature
- RCCA ejection
- Reactor coolant pump shaft seizure (locked rotor)

The partial loss of flow is less limiting than the complete loss of flow event; therefore only the complete loss of flow event is reanalyzed in the current reload safety evaluation. The startup of an inactive reactor coolant pump at an incorrect temperature is evaluated, rather than reanalyzed. The RCCA ejection and locked rotor events are reanalyzed for the TPBAR core design, including consideration of the higher initial fuel temperatures.

In addition to the events mentioned above, there are other non-LOCA transients that assume maximum fuel temperatures only to define a minimum value for the overall fuel-to-coolant heat transfer coefficient that is modeled in the LOFTRAN computer code (see Section 2.15.2.4). These events are:

- Loss of load/turbine trip
- Loss of normal feedwater
- Feedwater system pipe break (feedline break)
- Steamline break with coincident RCCA withdrawal at power

2.15.2.4 Computer Codes Used for the Non-LOCA Analysis

All of the computer codes used to perform analysis for the TPBAR core are also used in the safety analyses reported in the reference plant FSAR. The LOFTRAN computer code (Reference 7) is used to analyze the transient response of a pressurized water reactor system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing reactor vessel, hot and cold leg piping, steam generator (tube and shell side), and pressurizer. The pressurizer heaters, spray, and safety valves are also considered in the program. Point model neutron kinetics and the reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator uses a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The reactor protection system, certain control systems, and the emergency core cooling system are all modeled. LOFTRAN is a versatile program capable of performing accident evaluations, control system studies, and parameter sizing.

The FACTRAN computer code (Reference 8) 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 (pressure, flow, temperature, and density). The FACTRAN code includes a sufficiently large number of radial space increments to handle fast transients such as the rod ejection accident. Also included in the code are material properties that are functions of temperature and a

sophisticated fuel-to-clad gap heat transfer coefficient. FACTRAN includes the film boiling heat transfer correlations, zircaloy-water reaction modeling, and the ability to model partial melting of materials that is required to deal with post-DNB transients.

The TWINKLE computer code (Reference 9) is a multidimensional spatial neutron kinetics code, which is patterned after steady-state codes that have been used for reactor core design. 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 multiregion fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. Aside from basic cross-section data and thermal hydraulic parameters, the code accepts as input basic driving functions, such as inlet temperature, pressure, flow, boron concentration, control r l motion, and others. The TWINKLE code is used to predict the kinetic behavior of a ıctor for transients that cause a major perturbation in the spatial neutron flux distribution.

2.15.2.5 Increase In Heat Removal by the Secondary System

All of the accidents defined within this event category in the reference plant FSAR are non-LOCA transients that will be evaluated here with regard to impact of the TPBAR core design. Among the accidents in Section 15.1 of the reference plant FSAR, only the feedv ater system malfunction resulting in an increase in feedwater flow (15.1.2) requires actual analysis to assess the acceptability of the TPBAR core. That analysis follows. For the other events in the increase in heat removal by the secondary system category (listed below), a single evaluation will suffice to address the impact of the TPBAR reload core.

- 15.1.1 Feedwater System Malfunctions That Result in a Decrease in Feedwater Temperature
- 15.1.3 Excessive Increase in Secondary Steam Flow
- 15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve
- 15.1.5 Steam System Piping Failure

An evaluation has been performed to assess the potential impact, if any, of the TPBAR core design on the current reference plant licensing basis analyses for the non-LOCA events listed above. For all these events, the TPBAR core design has not changed any of the bounding values assumed for the key safety analysis parameters used in the reference plant FSAR analyses. Similarly, none of these events is affected by the increase in fuel temperatures. Therefore, the reference plant safety analysis for each of these non-LOCA events is unaffected by the TPBAR core design.

2.15.2.5.1 Feedwater System Malfunction Resulting in an Increase in Feedwater Flow (SRP 15.1.2)

Introduction

Section 15.1.2 of the reference plant FSAR addresses the effects of increase in feedwater flow (excessive feedwater) events initiated from both hot full power and hot zero power conditions.

The current reference plant FSAR licensing basis analysis for the event initiated from hot full power conditions is not affected by the TPBAR core design. For this case, the TPBAR core does not change any of the reference plant bounding values assumed for the key safety analysis parameters. Additionally, the previously discussed increase in maximum fuel temperatures has no effect on the results of the safety analysis for the excessive feedwater event initiated from hot full power conditions. Therefore, the conclusions of the reference plant FSAR remain valid for the excessive feedwater event initiated from hot full power conditions.

However, the excessive feedwater event initiated from hot zero power (HZP) conditions does require that some reanalysis be performed for the TPBAR reload evaluation. The current reference plant analysis for the HZP excessive feedwater accident relies on a bounding methodology that treats the event much like an uncontrolled RCCA withdrawal from subcritical (RWFS) accident. Since the RWFS event must be reanalyzed to address the change in the least-negative BOL Doppler-only power defect, it is also necessary to assess the HZP excessive feedwater case.

As described in the reference plant FSAR Section 15.1.2, the addition of excessive feedwater will produce a decrease in reactor coolant temperature which, in the presence of a negative moderator temperature coefficient of reactivity, causes an increase in core power. Such transients are restrained by the thermal capacity of the secondary plant and the RCS. Although not relied upon in the analysis to mitigate this transient, depending upon the initial power level, the high neutron flux trip, overpower delta-T (OPΔT) and overtemperature delta-T (OTΔT) trips may prevent a power increase which could lead to a DNBR less than the safety analysis limit value. Particularly for transient cases initiated from hot zero power conditions, the excessive feedwater event may produce a Safety Injection (SI) signal on the low steam pressure or low pressurizer pressure SI functions. An SI signal will terminate the event by generating a feedwater isolation signal.

Should no earlier protection function be actuated, continuous addition of excessive feedwater will ultimately be limited by the steam generator high-high level trip which closes all feedwater control and isolation valves, trips the main feedwater pumps, and trips the turbine. The increase in normal feedwater flow incident is classified as an ANS Condition II event, a fault of moderate frequency. The general acceptance criteria for this event category are discussed in Section 15.0.1.2 of the reference plant FSAR. The event is primarily analyzed to show that the reactor protection system is adequate to ensure that the DNB design basis continues to be met throughout the transient.

Method of Analysis

The current reference plant licensing basis analysis for the hot zero power excessive feedwater case uses the LOFTRAN computer code to define a bounding reactivity insertion rate that is then used in a TWINKLE code model to compute the core transient behavior. For the TPBAR reload analysis, a revised standard methodology is used in which the core heat flux and reactor coolant system response to the cooldown produced by the excessive feedwater event are modeled using only the LOFTRAN computer code. The THINC detailed core thermal-hydraulic computer code is then used to determine if DNB occurs for the core transient conditions predicted by LOFTRAN. In verifying the overall conservatism of this method, the

power predictions of the LOFTRAN point kinetics model are confirmed by comparison with a detailed 3-D core analysis for the limiting conditions of the transient. This detailed core analysis explicitly models the hypothetical core configuration (that is, the most limiting stuck RCCA, nonuniform inlet temperatures, RCS pressure, and core flow) and directly evaluates the total reactivity feedback of the core, incl ng power and density redistribution in an integral fashion.

For the reload evaluation, the event is analyzed using the same conservative analysis input assumptions as in the current reference plant FSAR. These include the use of a conservatively large negative moderator temperature coefficient, and a conservatively small Doppler power feedback and delayed neutron fraction. The analysis is performed for the following two cases that are initiated with the reactor assumed to be just critical at zero load conditions:

1. A step increase in feedwater flow to 225% of full power nominal to one steam generator

2. A step increase in feedwater flow to 225% of full power nominal to all four steam generators

<u>Results</u>

The results show that the case with a step increase of 225% feedwater flow to all four steam generators is more limiting, though the return to power for this case is not large and DNBR design limits are not approached. The transient predictions are shown in Figures 2.15.2.5.1-1 through 2.15.2.5.1-4. The results show that the RCS cooldown produced by the excessive feedwater flow event causes a rise in the nuclear power due to the reactivity increase. The reactor power rise is limited by the Doppler power feedback, and the transient is terminated by feedwater isolation after reaching the low pressurizer pressure SI setpoint. The core heat flux and nuclear power are shown in Figures 2.15.2.5.1-1 and 2.15.2.5.1-2, respectively. The heat flux lags behind the nuclear power due to the fuel rod thermal time constant. Figure 2.15.2.5.1-3 shows the actual cooldown transient in the core, while Figure 2.15.2.5.1-4 shows the pressurizer pressure transient.

The minimum DNBR, which occurs around the time of peak heat flux, remains above the applicable safety analysis limit for the reference plant. Therefore the DNB design basis is met at all times during the event. The sequence of events for the excessive feedwater event initiated at HZP conditions is found in Table 2.15.2.5.1-1. Once feedwater isolation occurs, the reactor returns to a stable condition, at which time normal plant shutdown procedures can be followed.

<u>Conclusions</u>

The analysis for the TPC in the reference plant shows that in the event of a feedwater system malfunction resulting in an increase in feedwater flow at hot zero power conditions, the core and RCS are not adversely affected since the DNB design basis, as defined in Section 4.4 of the reference plant FSAR, continues to be met. Therefore, the presence of the TPBARs in the core does not alter the conclusions of the FSAR with respect to this event.

Table 2.15.2.5.1-1
Time Sequence of Events for Feedwater System Malfunction that Results in an Increase in Feedwater Flow from Hot Zero Power Conditions
225% Feedwater Flow to All Four SGs

Event	Time (sec.)
One main feedwater control valve fails fully open	0.0
Safety Injection System low pressurizer pressure Signal generated	20.4
Feedwater isolation occurs	27.4
Peak heat flux occurs	61.0

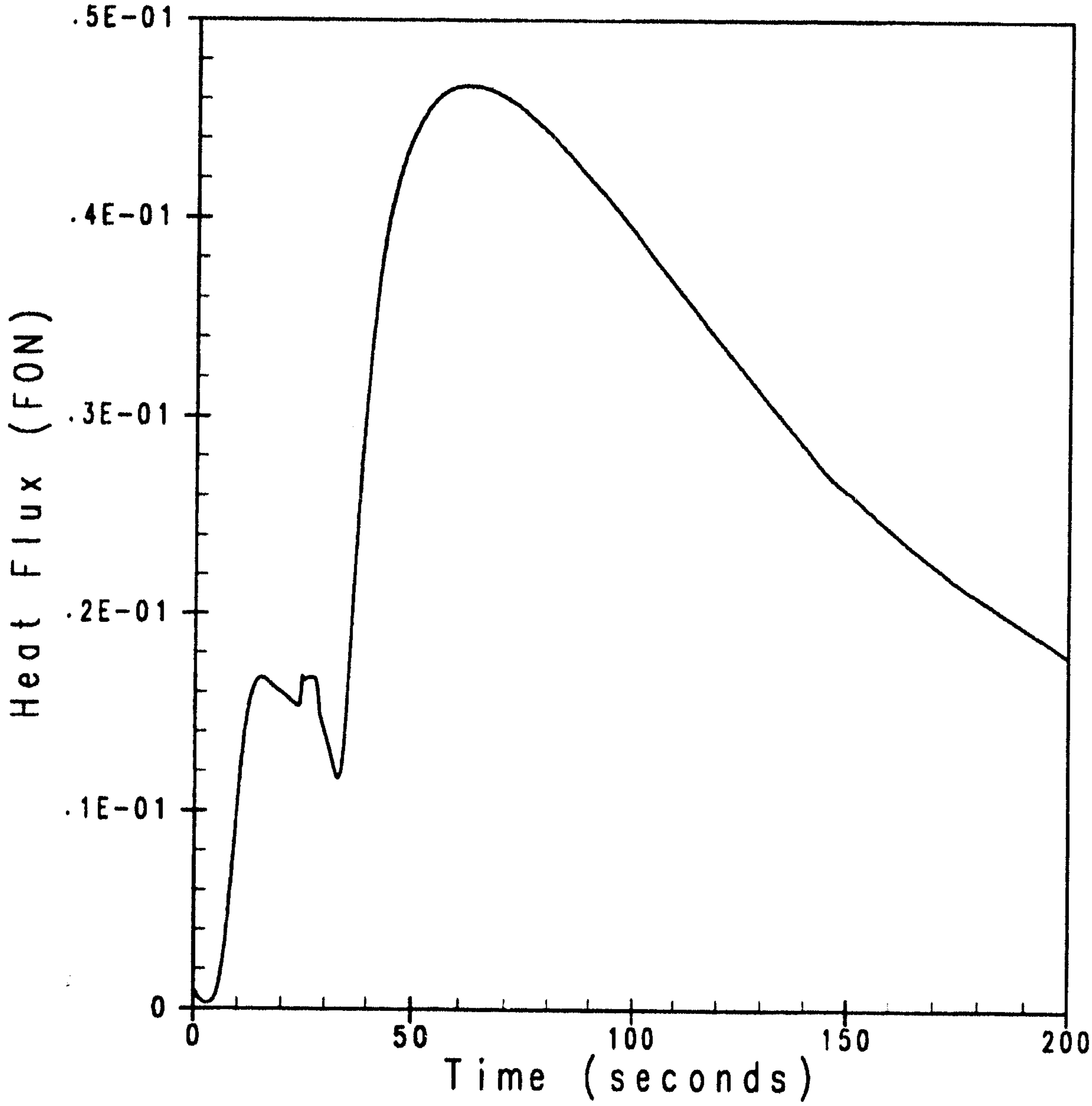


Figure 2.15.2.5.1-1
Heat Flux Transient for 225%
Feedwater Increase to All Four SGs

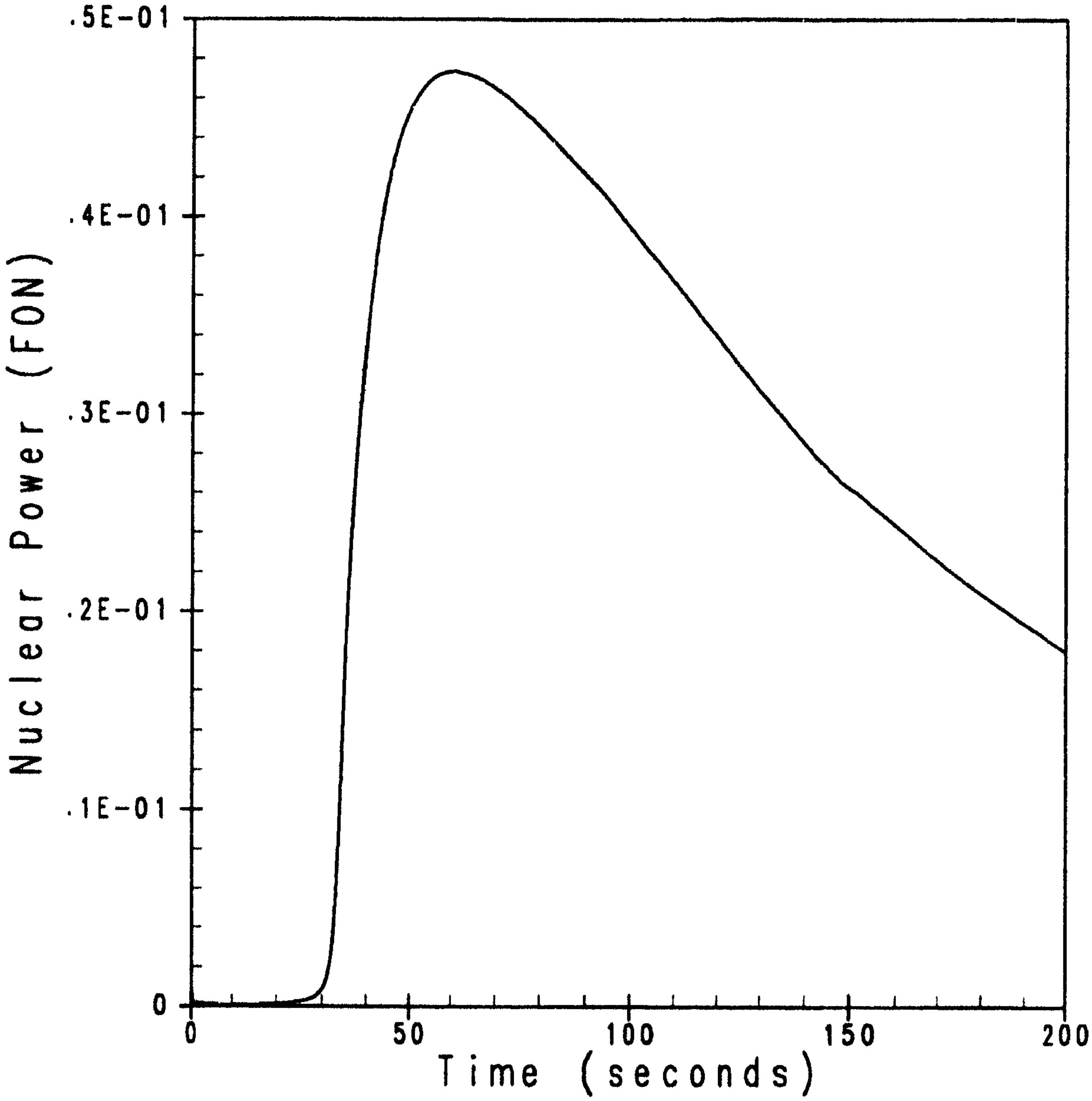


Figure 2.15.2.5.1-2
Nuclear Power Transient for 225%
Feedwater Increase to All Four SGs

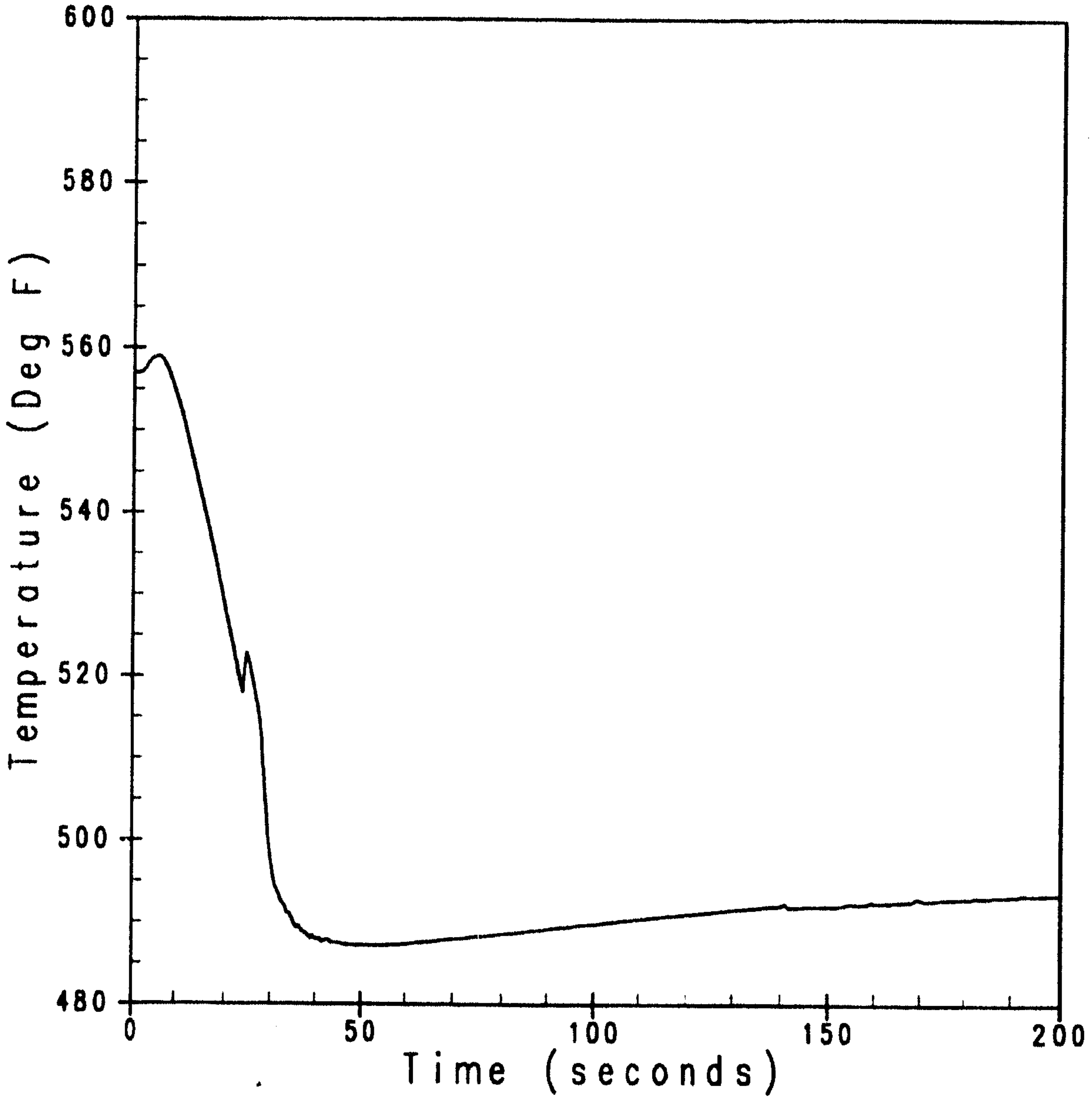


Figure 2.15.2.5.1-3
Average Coolant Temperature Transient for 225% Feedwater Increase to All Four SGs

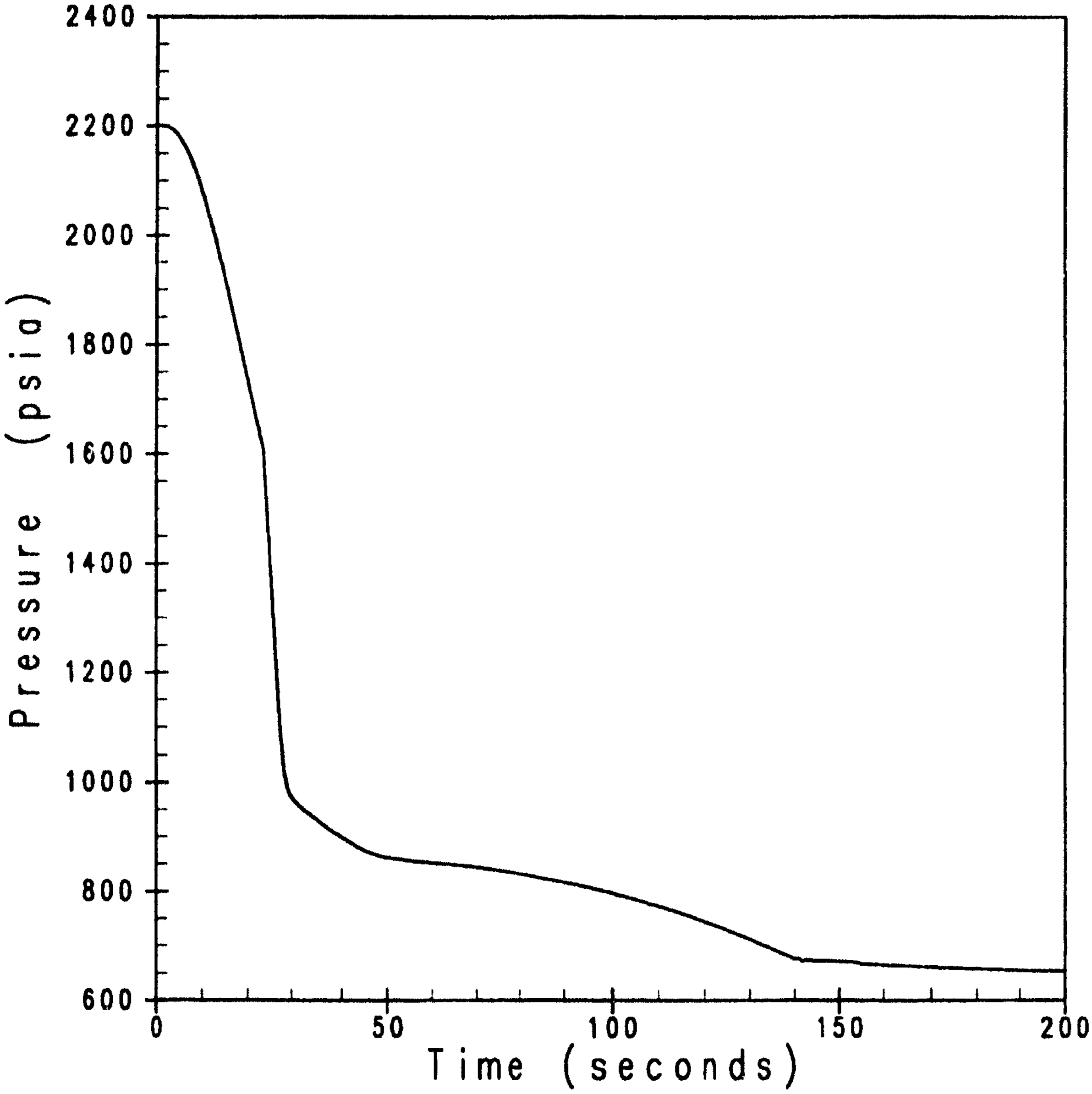


Figure 2.15.2.5.1-4
Pressurizer Pressure Transient for 225%
Feedwater Increase to All Four SGs

2.15.2.6 Decrease In Heat Removal by the Secondary System

All of the accidents analyzed within this event category in the reference plant FSAR are non-LOCA transients that will be evaluated here with regard to impact of the TPBAR core design. The FSAR does include the identification of an event in FSAR Section 15.2.1, steam pressure regulator malfunction or failure that results in decreasing steam flow, that is not applicable to the reference plant. For the other seven events within FSAR Section 15.2 that do apply to the reference plant, no actual reanalysis is required to address TPBAR core effects; an evaluation is sufficient. The list of non-LOCA decrease in heat removal by the secondary system events found in the reference plant FSAR is as follows:

15.2.2	Loss of External Electrical Load
15.2.3	Turbine Trip
15.2.4	Inadvertent Closure of Main Steam Isolation Valves
15.2.5	Loss of Condenser Vacuum & Other Events Resulting in Turbine Trip
15.2.6	Loss of Non-Emergency AC Power to the Plant Auxiliaries
15.2.7	Loss of Normal Feedwater Flow
15.2.8	Feedwater System Pipe Break

An evaluation has been performed to assess the potential impact, if any, of the TPBAR core design on the analyses for the non-LOCA events listed above. For all these events, the presence of TPBARs in the core design does not change any of the bounding values assumed for the key safety analysis parameters in the reference plant FSAR analyses. Thus, the TPBARs have no direct effect on the results for any of the events listed above.

As previously discussed, the current TPBAR core evaluation does include consideration of slightly increased maximum fuel temperatures, though those increases are not directly attributable to the TPBARs. The safety analyses for all of the FSAR Section 15.2 events listed above model maximum fuel temperatures as a bounding assumption, so the potential impact of increases in the bounding temperatures must be addressed.

The maximum fuel temperatures in question are used to define minimum values for the overall fuel to coolant heat transfer coefficients that are modeled in the LOFTRAN computer code. Sensitivity studies have demonstrated that these events are not significantly affected by much larger changes in the subject heat transfer coefficients than are associated with the fuel temperature increases considered for the TPBAR core. The slightly increased maximum fuel temperatures considered for the TPBAR core do not significantly affect the results for any of the decrease in heat removal events considered in Section 15.2 of the reference plant FSAR. Therefore, all safety analysis criteria for the non-LOCA decrease in heat removal events are met by the TPBAR core design.

2.15.2.7 Decrease In Reactor Coolant System Flow Rate

All of the accidents defined within this event category in the reference plant FSAR are non-LOCA transients that will be evaluated here with regard to impact of the TPBAR core design. The list of decrease in reactor coolant flow rate events found in the reference plant FSAR is as follows:

15.3.1 Partial Loss of Forced Reactor Coolant Flow

15.3.2 Complete Loss of Forced Reactor Coolant Flow

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

15.3.4 Reactor Coolant Pump Shaft Break

For all these events, the presence of the TPBARs in the core does not change any of the bounding values assumed for the key safety analysis parameters in the reference plant FSAR analyses. Thus the TPBARs have no direct effect on the results for any of the events listed above.

As previously discussed, the current TPBAR core evaluation does include consideration of slightly increased maximum fuel temperatures, though those increases are not directly attributable to the TPBARs. The safety analyses that address all of the events listed above model maximum fuel temperatures as a bounding assumption, so the potential impact of increases in the bounding temperatures must be addressed. In fact, the analyses for these events directly predict fuel rod response using the FACTRAN computer code. Therefore, the results for these events are potentially more sensitive to changes in the assumed fuel temperatures than events modeled using the LOFTRAN computer code, which does not use fuel temperature as a direct input. As a result, actual reanalysis for the limiting cases has been performed to assess the impact of the TPBAR core on the events considering a decrease in the reactor coolant system flow.

The results for the partial loss of forced reactor coolant flow event are bounded by those for the complete loss of forced reactor coolant flow. Therefore, in support of the current evaluation, formal reanalysis has been performed for the complete loss of forced reactor coolant flow, but not for the partial loss of reactor coolant flow event. Similarly, as indicated in Section 15.4 of the reference plant FSAR, the results for the locked rotor event bound those for the reactor coolant pump shaft break. As a result, the current TPBAR core evaluation includes reanalysis for the locked rotor event, but none for the shaft break accident.

2.15.2.7.1 Partial Loss of Forced Reactor Coolant Flow (SRP 15.3.1)

The consequences of the partial loss of forced reactor coolant flow event are bounded by those for the complete loss of forced reactor coolant flow accident. The results for the complete loss of forced reactor coolant flow are presented in Section 2.15.2.7.2 of this report.

2.15.2.7.2 Complete Loss of Forced Reactor Coolant Flow (SRP 15.3.2)

Introduction

Section 15.3.2 of the reference plant FSAR describes the complete loss of forced reactor coolant flow event, which may result from a simultaneous loss of electrical power to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of a loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent adverse effects on the fuel if the reactor were not tripped promptly.

The complete loss of forced reactor coolant flow event is classified as an ANS Condition III event, i.e., an infrequent fault. The general criteria for this event category are discussed in Section 15.0.1.3 of the reference plant FSAR. The actual limiting criterion that is conservatively applied to this event by Westinghouse is to demonstrate that the DNB design basis is met.

The signals which provide the necessary protection for this event at the reference plant are:

1. Reactor coolant pump power supply bus undervoltage
2. Reactor coolant pump power supply bus underfrequency
3. Low reactor coolant loop flow

These trip functions are fully described in the FSAR. For the analysis of the complete loss-of-coolant flow event, the reactor trip actually assumed in the analysis is the reactor coolant pump bus undervoltage trip.

Method of Analysis

The general method of analysis and the assumptions made regarding initial operating conditions are identical to those discussed in the reference plant FSAR. The need to reanalyze the complete loss of reactor coolant flow event is not strictly related to the presence of TPBARs in the core. Rather, the analysis is performed to assess the impact of the slightly higher (maximum increase of 21°F) initial fuel temperatures being modeled for the TPBAR reload core.

The complete loss of flow event is analyzed using a +7.0 pcm/°F full power moderator temperature coefficient (MTC). This is overly conservative since the reference plant technical specifications require that the MTC not exceed 0 pcm/°F at full power, though a higher MTC is allowed when operating at lower power levels.

Results

Figures 2.15.2.7.2-1 through 2.15.2.7.2-3 show the transient response for the loss of power to all RCPs with the assumed trip on the reactor coolant pump bus undervoltage function. Figure 2.15.2.7.2-1 shows the predicted RCS flow coastdown, while Figures 2.15.2.7.2-2 and 2.15.2.7.2-3 present the nuclear power and pressurizer pressure transients, respectively. An assessment of the limiting thermal hydraulic statepoint generated for this transient has confirmed that the DNB design basis continues to be met throughout the duration of the event. Since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is

not significantly reduced. Thus, the average fuel and clad temperature do not increase significantly (increases are less than 15°F) above their respective initial values. The calculated time sequence of events for the analyzed complete loss of forced reactor coolant flow case is shown in Table 2.15.2.7.2-1.

Conclusions

The analysis shows that the DNB design basis, as described in Section 4.4 of the reference plant FSAR, is met at all times during the transient. Thus, the analysis does not predict any adverse fuel effects or clad rupture and all applicable acceptance criteria are met for the TPBAR reload core under consideration. Therefore, the conclusions presented in the reference plant FSAR, for this event, remain valid for the TPBAR core design.

Table 2.15.2.7.2-1
Time Sequence of Events for Complete Loss of Forced Reactor Coolant Flow

Event	Time (sec.)
All operating pumps lose power and begin coasting down	0.0
Reactor coolant pump bus undervoltage trip setpoint reached	0.0
Rods begin to drop	1.5
Minimum DNBR occurs	3.2

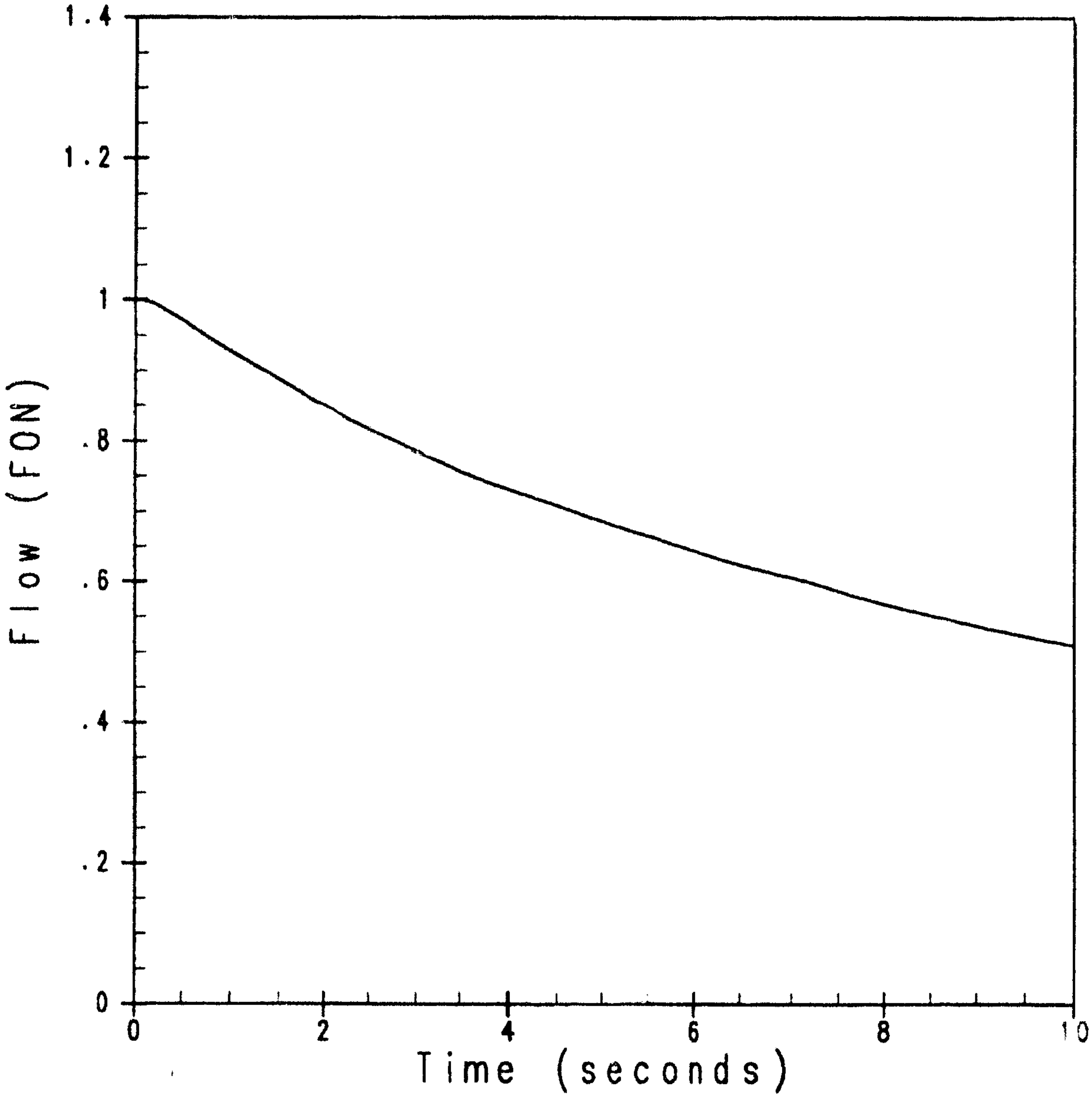


Figure 2.15.2.7.2-1
RCS Flow Transient for Four Loops In Operation,
Four Pumps Coasting Down

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

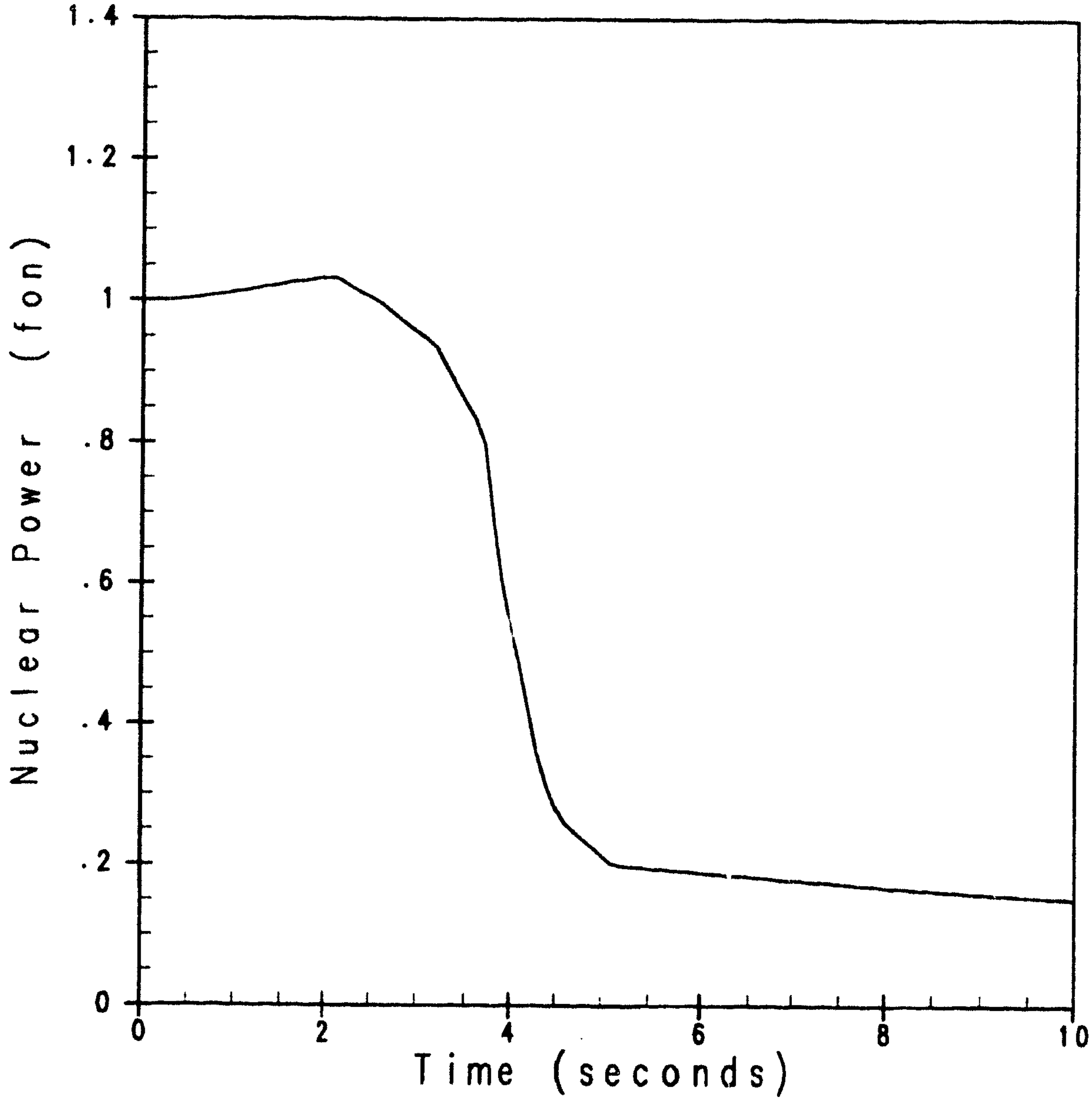


Figure 2.15.2.7.2-2
Nuclear Power Transient for Four Loops In Operation, Four Pumps Coasting Down

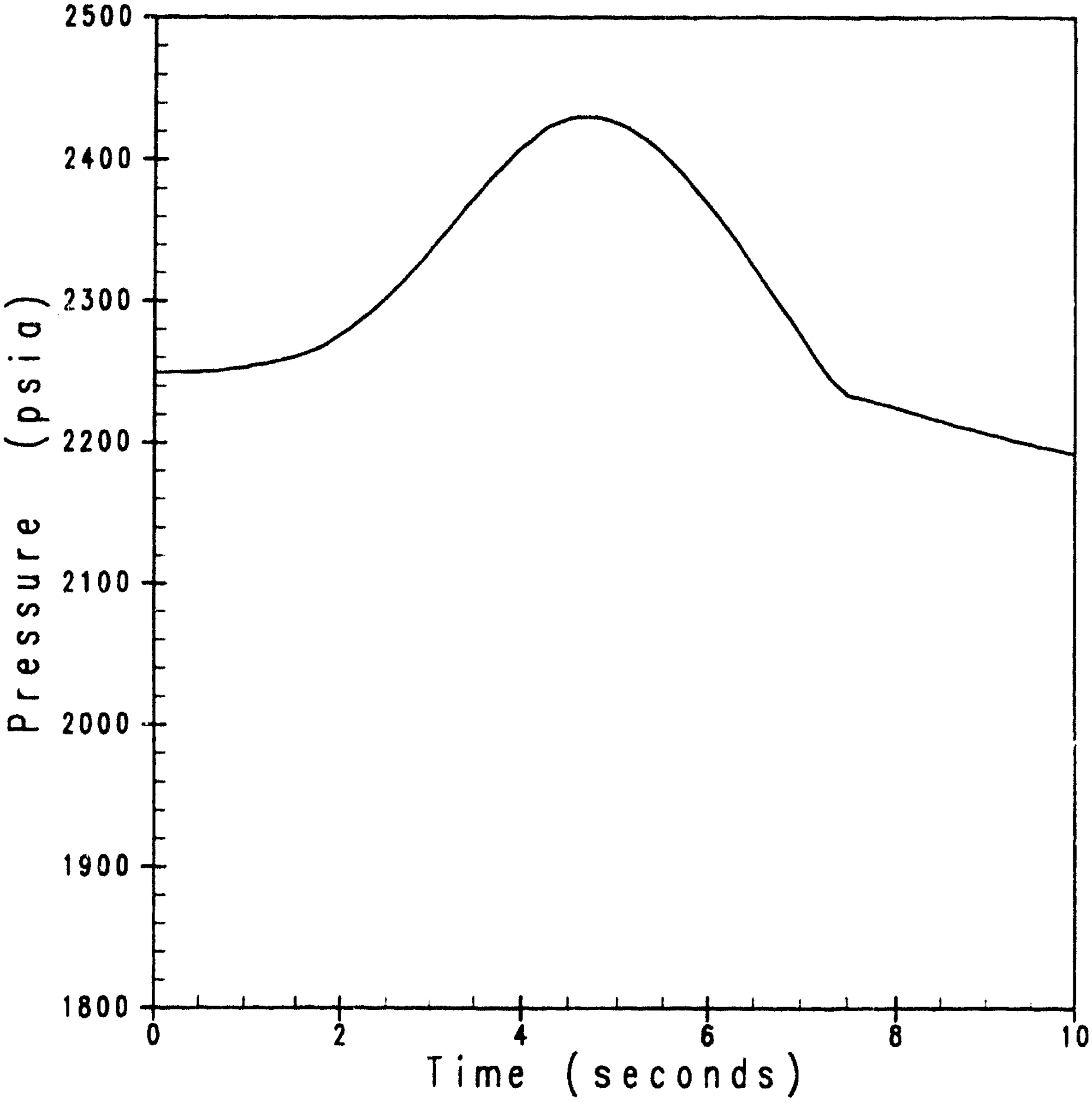


Figure 2.15.2.7.2-3
Pressurizer Pressure Transient for Four Loops In Operation, Four Pumps Coasting Down

2.15.2.7.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor) (SRP 15.3.3)

Introduction

The reactor coolant pump shaft seizure (locked rotor) event is discussed in Section 15.3.3 of the reference plant FSAR. For the instantaneous seizure of a reactor coolant pump rotor, flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a low flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generator in the loop with the locked rotor is reduced, first because the reduced reactor coolant system flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer in the steam generator causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system.

The insurge into the pressurizer causes a reactor coolant system pressure increase which in turn actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves in a sequence dependent on the rate of insurge and pressure increase. The power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident; however, for conservatism, their pressure-reducing effect as well as the pressure-reducing effect of the spray are not included in this analysis.

The locked rotor event is classified as an ANS Condition IV incident, i.e., a limiting fault. The general acceptance criteria applicable to this event category are discussed in Section 15.0.1.4 of the reference plant FSAR. The specific limiting criteria applicable to this event are as discussed in Section 15.3.3 of the reference plant FSAR. They include the requirement that, at no time during the transient, can the reactor coolant system pressure exceed that which would causes stresses to exceed the faulted condition stress limits of the ASME Code, Section III. Also, the reference plant analysis for this event includes the calculation of the percentage of fuel rods postulated to experience DNB. This value is used as input to the consideration of the radiological consequences associated with this event.

Method of Analysis

The general methodology used for the current analysis of this event is consistent with the reference plant FSAR. As in the FSAR, the analysis for this reload safety evaluation includes consideration of the pressure transient, the consequences of the assumed DNB, a conservative film boiling coefficient, a fuel clad gap coefficient that maximizes clad temperature, and the zirconium steam reaction. Although not strictly related to the presence of the TPBARs in the core, the current analysis includes consideration of slightly higher (maximum increase of 21°F) initial fuel temperatures in the TPBAR core.

The locked rotor analysis performed to evaluate fuel rod conditions at the hot spot and the reactor coolant system pressure assumes a +7.0 pcm/°F full power moderator temperature coefficient (MTC). This is overly conservative since the reference plant technical specifications require that the MTC not exceed 0.0 pcm/°F at full power. The analysis performed to assess the fraction of fuel rods that experience DNB during the transient assumes that the MTC is at

0.0 pcm/°F, which is consistent with the reference plant technical specifications. Both the MTC values assumed in the current analysis are consistent with the assumptions used in Section 15.3.3 of the reference plant FSAR for the locked rotor event.

The reference plant FSAR reports results for locked rotor cases both with and without a post-trip loss of offsite power. The current TPBAR reload evaluation presents results for only the case with a loss of offsite power, since that case is slightly more limiting . Power is assumed to be lost to the unfaulted reactor coolant pumps 2.0 seconds after reactor trip. This 2.0 second delay is a conservative assumption based on grid stability analyses.

Also, consistent with the analysis for this event reported in the reference plant FSAR, the current cases analyzed for fuel hot spot and reactor coolant system overpressure concerns use the Westinghouse Standard Thermal Design Procedure (STDP) to define the initial condition assumptions. In this methodology, initial conditions are obtained by applying the maximum steady state errors to rated values. In contrast, the locked rotor rods in DNB analysis assumes initial conditions as defined by the Westinghouse Revised Thermal Design Procedure (RTDP, Reference 10), which involves a statistical approach to defining the initial conditions associated with certain analyses that address the potential onset of DNB.

Results

The transient results for the locked rotor accident with a loss of offsite power, are shown in Figures 2.15.2.7.3-1 through 2.15.2.7.3-5. These results are for the case analyzed to evaluate the reactor coolant system pressure transient and the fuel rod hot spot response to the locked rotor event. The calculated sequence of events for this case is shown in Table 2.15.2.7.3-1, while the results are summarized in Table 2.15.2.7.3-2. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits of the ASME Code, Section III. Also the peak clad temperature remains well below design limits. It should be noted that the clad temperature is conservatively calculated assuming that DNB occurs at the initiation of the transient.

The conservative locked rotor analysis performed for the TPBAR reload evaluation confirms that the fraction of the fuel rods predicted to undergo DNB continues to be bounded by the value reported in the reference plant FSAR. That is, less than 5% of the fuel rods are predicted to undergo DNB and are assumed to release their gap gas inventory to the reactor as input to the dose calculations.

Conclusions

For the locked rotor event, the peak reactor coolant system pressure predicted for the TPBAR core is less than that which would cause stresses to exceed the faulted condition stress limits, thereby assuring that the integrity of the primary coolant system is maintained. Similarly, the peak clad temperature calculated for the hot spot remains considerably less than 2700°F, so the core will remain in place and intact with no loss of core cooling capability. A 2700°F clad temperature limit has been used by Westinghouse as a conservative criterion to ensure continued core coolability during the short duration locked rotor event. The percentage of fuel rods in the core postulated to undergo DNB during the locked rotor event continues to be less than 5%, which is consistent with the results reported in the reference plant FSAR. Therefore,

the conclusions presented in the reference plant FSAR, for this event, remain valid for the TPBAR core design.

Table 2.15.2.7.3-1
Time Sequence of Events for Reactor Coolant Pump Shaft Seizure (Locked Rotor)
Hot Spot Case With Loss of Offsite Power

Event	Time (sec.)
Rotor on one RCP locks	0.0
Low flow trip setpoint reached	0.02
Rods begin to drop	1.02
Loss of offsite power-three unfaulted RCPs begin flow coastdown	3.02
Maximum RCS pressure occurs	3.40
Maximum clad average temperature occurs	3.59

Table 2.15.2.7.3-2 Summary of Results for Locked Rotor Transient Hot Spot Case With Loss of Offsite Power	
Maximum RCS pressure	2670 psia
Maximum clad average temperature at core hot spot	2059°F
Zr-H_2O reaction at core hot spot	0.75 w/%

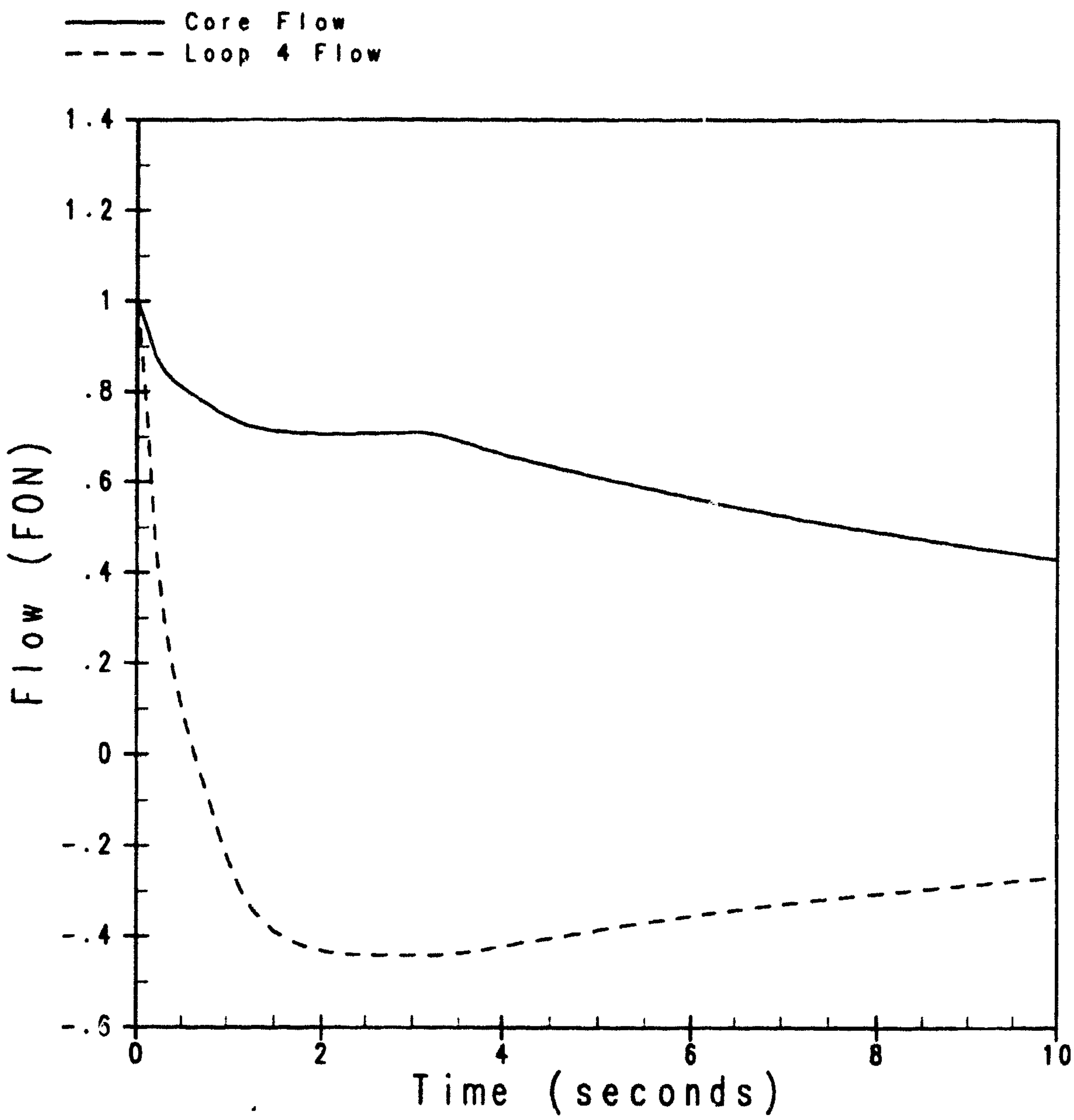


Figure 2.15.2.7.3-1
Flow Transients for Four Loops In Operation
(One Locked Rotor Without Offsite Power)

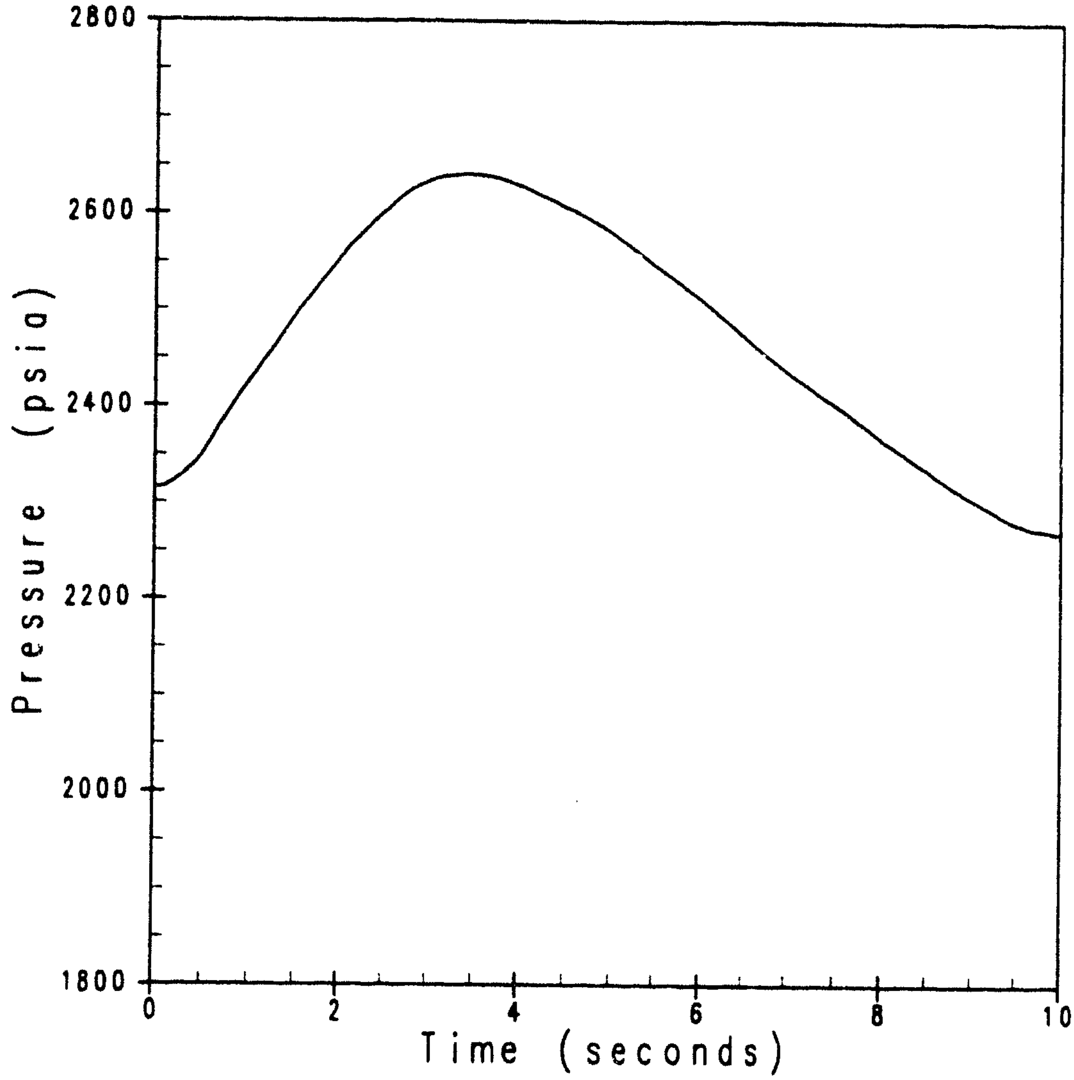


Figure 2.15.2.7.3-2
Peak Reactor Coolant Pressure Transient for Four Loops In Operation
(One Locked Rotor Without Offsite Power)

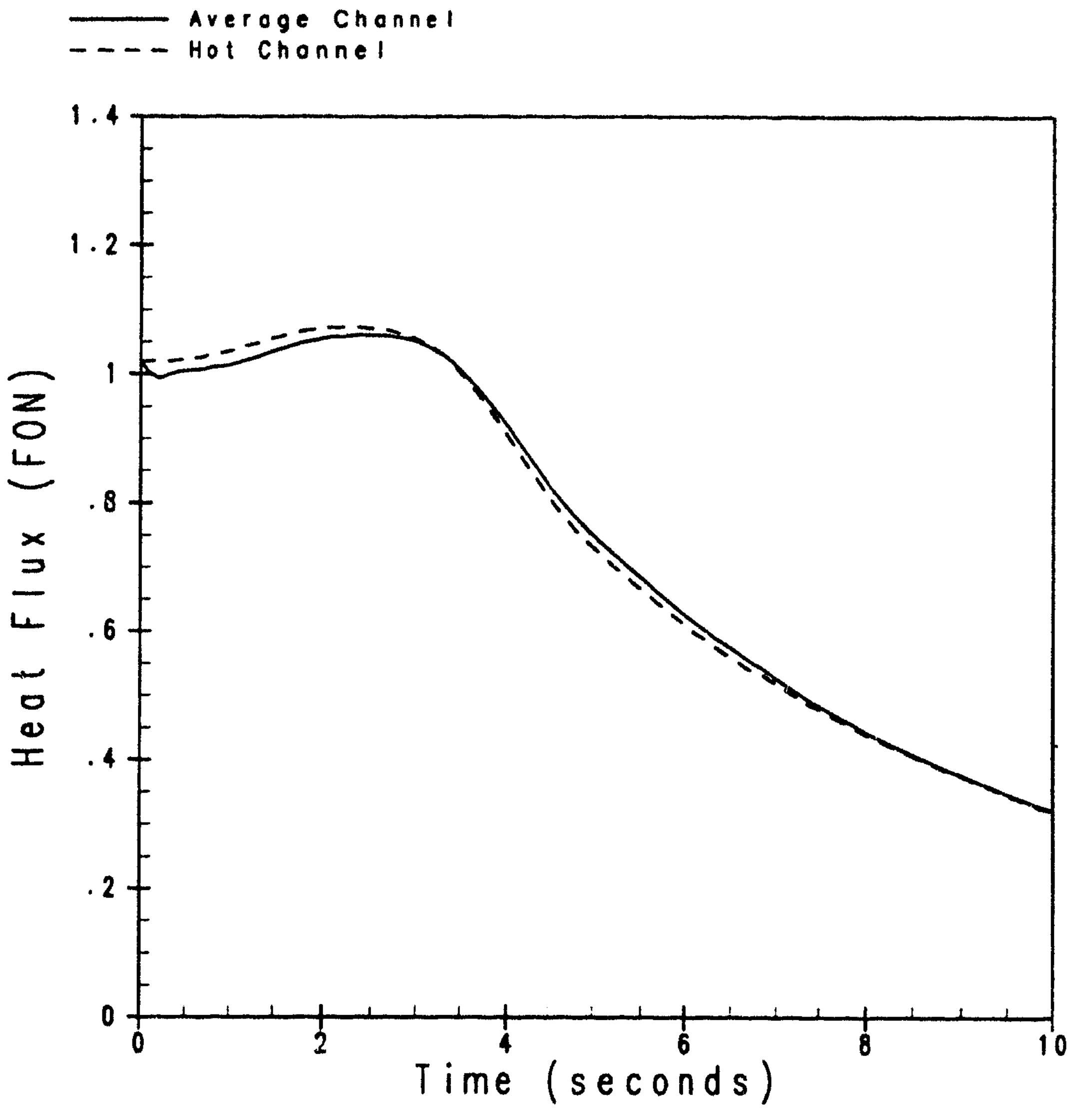


Figure 2.15.2.7.3-3
Heat Flux Transients for Four Loops in Operation
(One Locked Rotor Without Offsite Power)

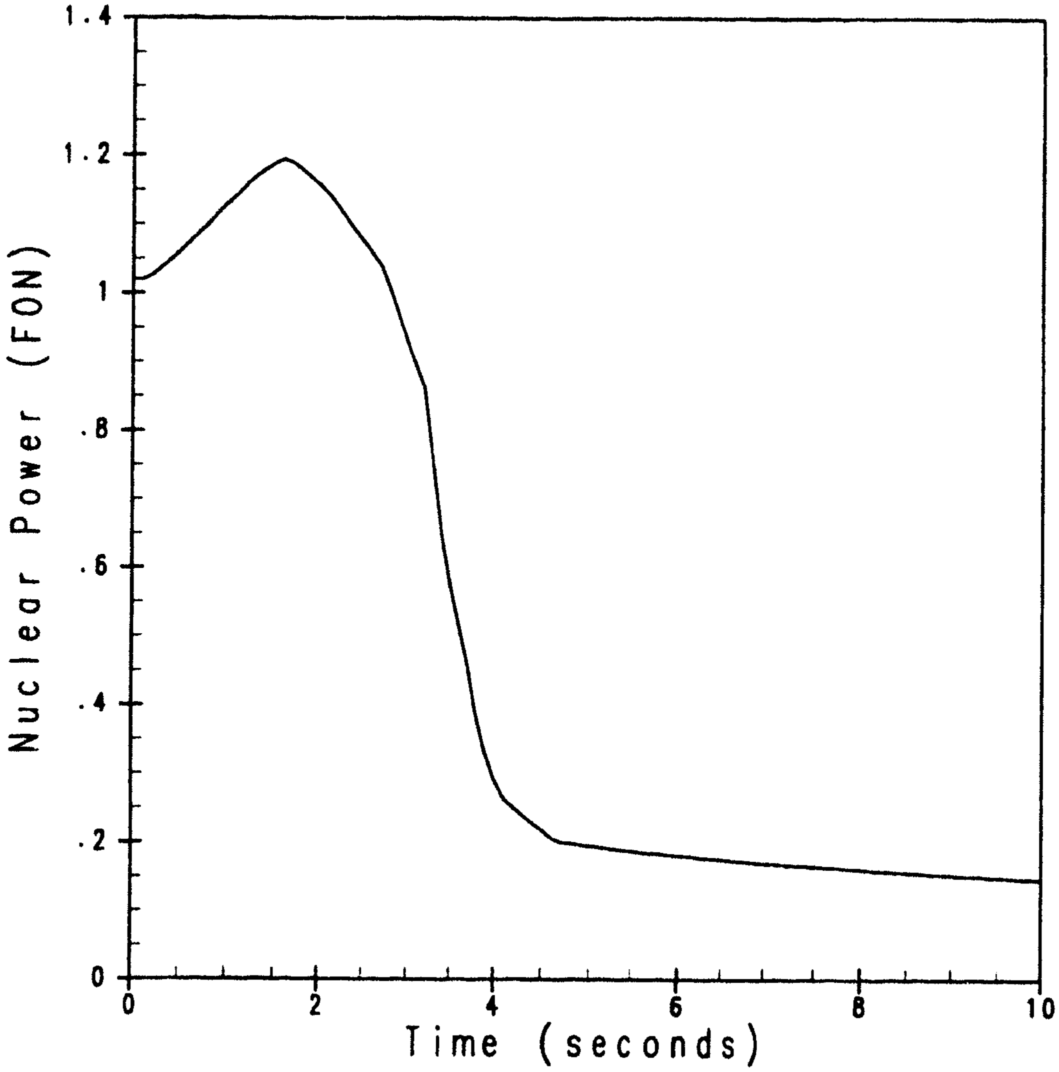


Figure 2.15.2.7.3-4
Nuclear Power Transient for Four Loops in Operation
(One Locked Rotor Without Offsite Power)

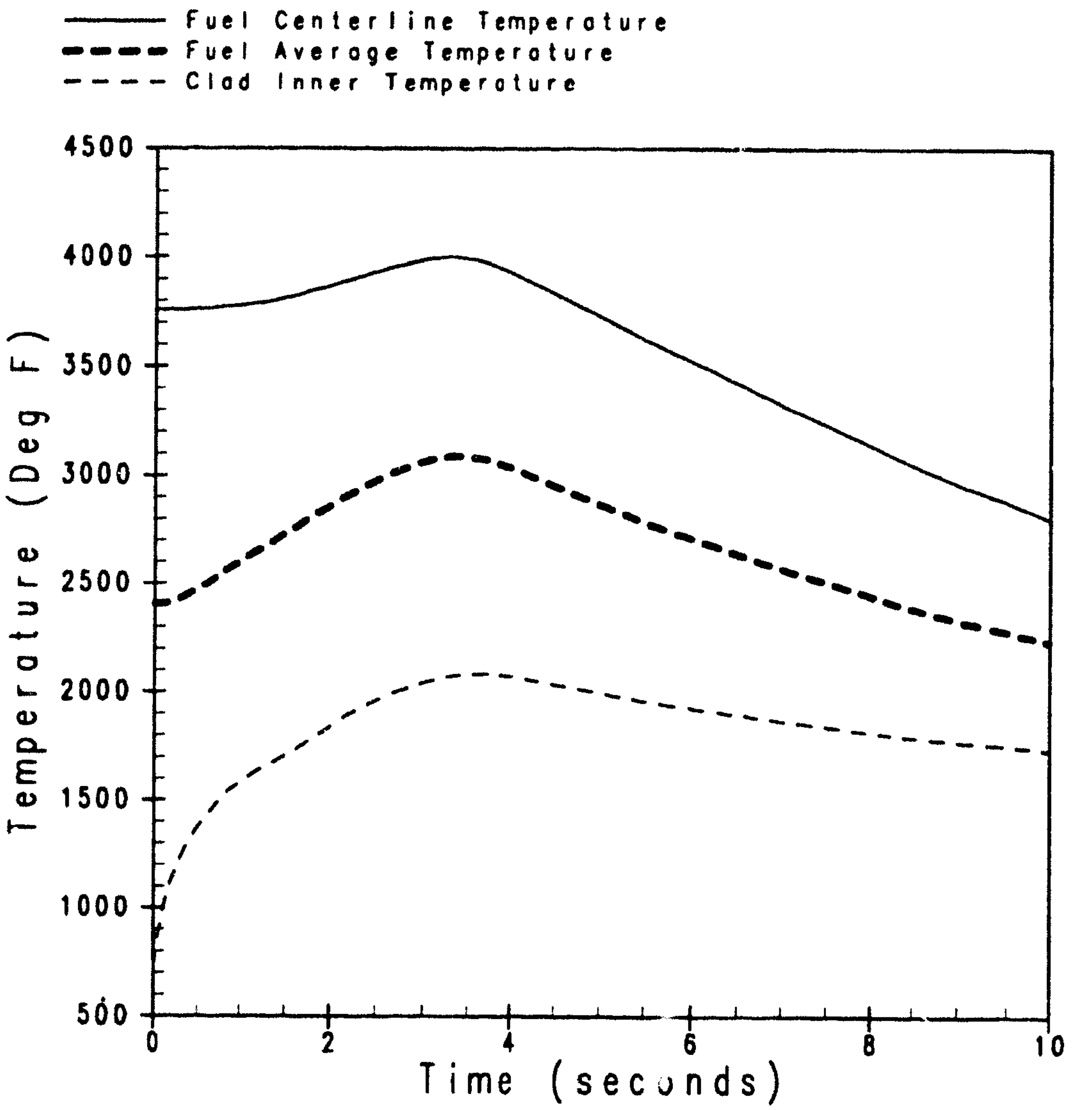


Figure 2.15.2.7.3-5
Fuel Temperature Transients for Four Loops In Operation
(One Locked Rotor Without Offsite Power)

2.15.2.7.4 Reactor Coolant Pump Shaft Break (SRP 15.3.4)

The results for the analysis of the locked rotor event, found in Section 2.15.2.7.3 of this report are more limiting than those for the reactor coolant pump shaft break event. With a failed shaft, the impeller could conceivably be free to spin in a reverse direction as opposed to being fixed in position as would occur with a locked rotor. During that portion of the transient with reverse flow in the broken shaft loop, a free spinning shaft would reduce the amount of core flow. However, the assumptions in the locked rotor analysis conservatively bound the broken shaft case by modeling the resistance of a locked rotor during forward flow in the faulted loop, but the resistance of a broken shaft during reverse flow in that loop. Based on the conclusions for the locked rotor event, it can be stated that acceptable results will be obtained for the reactor coolant shaft break event with the TPBAR core.

2.15.2.8 Reactivity and Power Distribution Anomalies

All but one of the accidents actually analyzed within this event category in the reference plant FSAR are non-LOCA transients that will be evaluated here with regard to impact of the TPBAR core design. The FSAR does identify an event in FSAR Section 15.4.5, a malfunction or failure of the flow controller in a boiling water reactor loop that results in an increased reactor flow rate, that is not applicable to the reference plant. Section 15.4.7 of the reference plant FSAR documents the inadvertent loading and operation of a fuel assembly in an improper position event, that is analyzed on a strictly steady state basis using nuclear design codes and methods without any direct non-LOCA transient input. For the other seven events within FSAR Section 15.4 that do apply to the reference plant, actual reanalysis is required for two accidents to address TPBAR core effects. Evaluations are sufficient for the other five events. The complete list of non-LOCA events found in Section 15.4 of the reference plant FSAR is as follows:

15.4.1 Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition

15.4.2 Uncontrolled RCCA Bank Withdrawal at Power

15.4.3 RCCA Misalignment (Including Drop)

15.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

15.4.6 Chemical and Volume Control System Malfunctions that Result in a Decrease in the Boron Concentration in the Reactor Coolant (Boron Dilution)

15.4.8 Spectrum of RCCA Ejection Accidents

15.4.9 Steamline Break With Coincident RCCA Withdrawal at Power

The presence of the TPBARs in the core does result in a change to one of the key nuclear design related bounding parameters that is assumed as input to the safety analysis for the uncontrolled RCCA bank withdrawal from subcritical and RCCA ejection events. The analyses for both of these accidents use the least-negative Doppler-only power defect at BOL as direct input. As

indicated in Section 2.15.2.3 of this report, that input to the analysis has been reduced from 998 to 920 pcm. Also, the RCCA ejection cases initiated from hot full power directly model fuel rod performance using the FACTRAN computer code, so that they may be impacted by the increase in the maximum fuel temperatures. Therefore, reanalysis has been performed for both of these events to assess the impact of the TPBAR core, and those analyses are documented below.

The analyses for the startup of an inactive reactor coolant pump at an incorrect temperature event and the steamline break with coincident RCCA withdrawal at power event both model initial conditions that assume maximum fuel temperatures. Therefore they may be affected by the increase in maximum fuel temperatures that is considered in this TPBAR core reload evaluation. The presence of the TPBARs in the core does not directly affect any of the key safety analysis parameters assumed for these two events. For each of these events an evaluation is provided below to address the impact, if any, of the increase in maximum fuel temperatures.

For the other three non-LOCA events in the reactivity and power distribution anomaly category (RCCA bank withdrawal at power, RCCA misalignment, and boron dilution), a single evaluation is able to adequately assess the potential impact, if any, of the TPBAR core design on the associated safety analyses. For these events, the presence of TPBARs in the core design does not change any of the bounding values assumed for the key safety analysis parameters in the reference plant FSAR analyses. Additionally, none of these three events model maximum fuel temperatures, so the increase in those temperatures considered in the current TPBAR evaluation has no effect on these events. Thus, the TPBAR core design does not affect the nature of the predicted transients or the results for these three non-LOCA events.

2.15.2.8.1 Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition (SRP 15.4.1)

Introduction

As described in Section 15.4.1 of the reference plant FSAR, an RCCA withdrawal from subcritical event is defined as an uncontrolled addition of reactivity to the reactor core caused by the withdrawal of RCCA banks resulting in a power excursion. While the occurrence of a transient of this type is highly unlikely, such a transient could be caused by a malfunction of the reactor control or rod control rod systems. This event is classified as an ANS Condition II event, i.e. a fault of moderate frequency. The general acceptance criteria for this event category are discussed in Section 15.0.1.2 of the reference plant FSAR. The specific limiting acceptance criterion applied to this event is to demonstrate that the reactor protection system is adequate to preclude a violation of the DNB design basis.

As discussed more fully in Section 15.4.1 of the reference plant FSAR, should a continuous RCCA withdrawal accident occur, the automatic features of the reactor protection system available to terminate the transient are as follows:

1. Source range high neutron flux reactor trip
2. Intermediate range high neutron flux reactor trip
3. Power range high neutron flux reactor trip (low setting)

4. Power range high neutron flux reactor trip (high setting)
5. High nuclear flux rate reactor trip

In addition, control rod stops on high intermediate range flux and high power range flux serve to discontinue rod withdrawal and may preclude the need to actuate the intermediate range or power range flux trips, respectively.

Method of Analysis

The uncontrolled RCCA bank withdrawal from subcritical accident for the TPBAR core is reanalyzed following the same basic methodology and using the same computer codes as described in Section 15.4.1.2 of the reference plant FSAR. The analysis is performed in three stages: First, an average core nuclear power transient calculation, then an average core heat transfer calculation, and finally the DNBR calculation. For DNBR calculations downstream of the mixing vane grids, the WRB-2 correlation is applied. However, because this transient produces a limiting core axial power distribution that is severely peaked to the bottom of the core, the DNBR is also calculated in the first grid span, which is downstream from a non-mixing vane grid. Due to absence of a mixing vane grid, the WRB-2 correlation is not applicable in this lower grid span. Here the W-3 correlation is used to evaluate DNBR.

The general assumptions indicated in the reference plant FSAR for this event continue to apply, although certain analysis parameters are modified to reflect the TPBAR core design. Selected key analysis assumptions are as follows:

1. A conservatively low value for the Doppler zero to full power reactivity defect. A value of -0.92% Δk which is specific to the TPBAR core design is used.

2. The analysis assumes a bounding MTC of +7 pcm/°F at the hot zero power initial conditions, which is consistent with the reference plant technical specifications.

3. The reactor is assumed to be at hot zero power coolant conditions (557°F). As discussed in the reference plant FSAR, this assumption is more conservative than that of a lower initial system temperature.

4. Consistent with the reference plant FSAR, reactor trip for this event is initiated by the power range high neutron flux (low setting) function. A safety analysis trip point of 35% of nominal power is assumed, including a 10% error allowance. The reactor trip insertion characteristics assume that the highest worth rod is stuck in its fully-withdrawn position.

5. The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of two sequential control banks having the greatest combined worth at maximum speed (45 in/min). The most limiting axial and radial power shapes associated with this bank position are assumed in the DNB analysis.

6. The initial power level assumed (10^{9} of nominal) is below that expected for any shutdown condition. The combination of highest reactivity insertion rate and low initial power produces the highest peak heat flux.

7. Two out of the four reactor coolant pumps are assumed to be in operation, which is consistent with the reference plant technical specifications for Mode 3. The assumption of reduced initial reactor coolant system flow is conservative for DNB calculations.

Results

The nuclear power, heat flux, fuel average temperature and clad temperature as a function of time are shown in Figures 2.15.2.8.1-1 through 2.15.2.8.1-3. The period of peak heat flux is of a relatively short duration that limits the resulting energy release and associated fuel temperature increase. The thermal heat flux, which is a primary factor in determining the potential for DNB, remains well below the full power nominal value as a result of the inherent thermal lag in the fuel. A calculation of the minimum DNBR at the time of the peak heat flux confirms that the DNB design basis, as defined in Section 4.4 of the reference plant FSAR, is met for the TPBAR core.

The sequence of events for this transient is presented in Table 2.15.2.8.1-1. Once the reactor trip occurs, the reactor returns to a stable condition, at which time normal plant shutdown procedures can be followed.

Conclusions

The results show that in the event of an RCCA withdrawal from subcritical accident in the TPBAR core, the core and RCS are not adversely affected since the DNB design basis continues to be met. Therefore, the conclusions presented in the reference plant FSAR, for this event, remain valid.

Table 2.15.2.8.1-1
Time Sequence of Events for Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Condition

Event	Time (sec.)
Initiation of uncontrolled RCCA withdrawal from 10^{-9} of nominal power	0.00
Power range high neutron flux low setpoint reached	12.49
Peak nuclear power occurs	12.64
Rods begin to fall into core	12.99
Peak heat flux occurs	14.85
Peak clad average temperature occurs	15.13
Peak average fuel temperature occurs	15.33

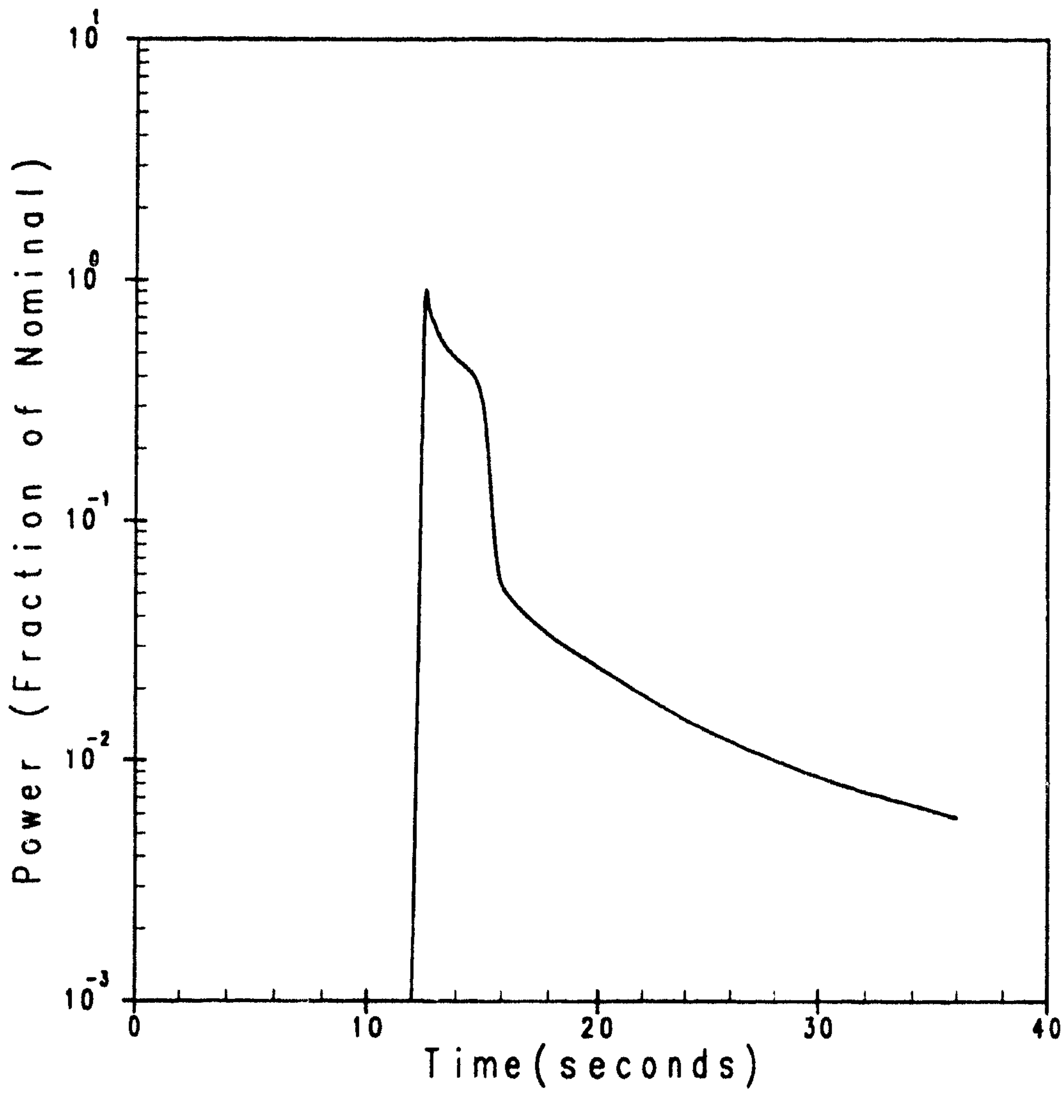


Figure 2.15.2.8.1-1
Nuclear Power Transient for Uncontrolled Rod Withdrawal From a Subcritical Condition

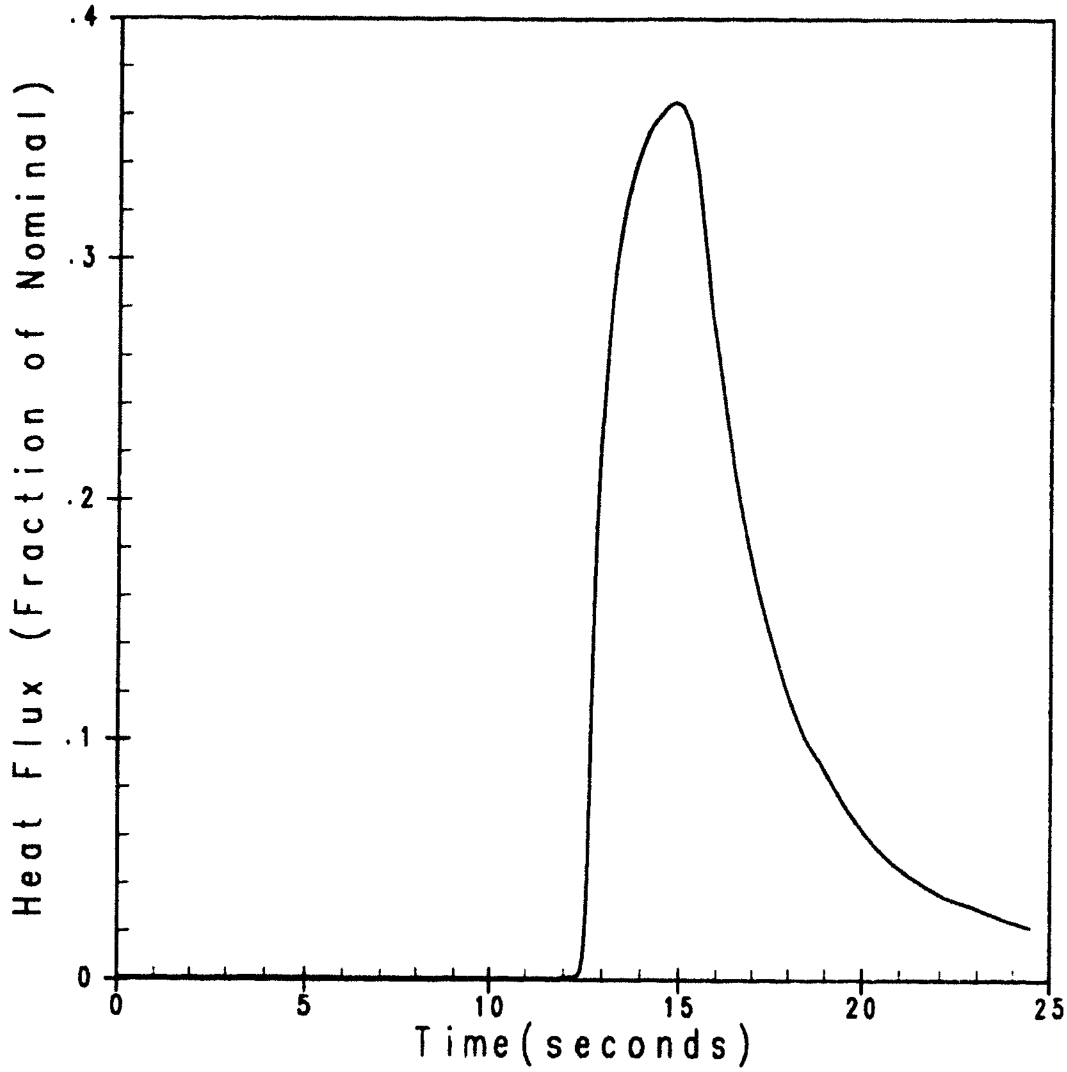


Figure 2.15.2.8.1-2
Heat Flux Transient for Uncontrolled Rod Withdrawal
From a Subcritical Condition

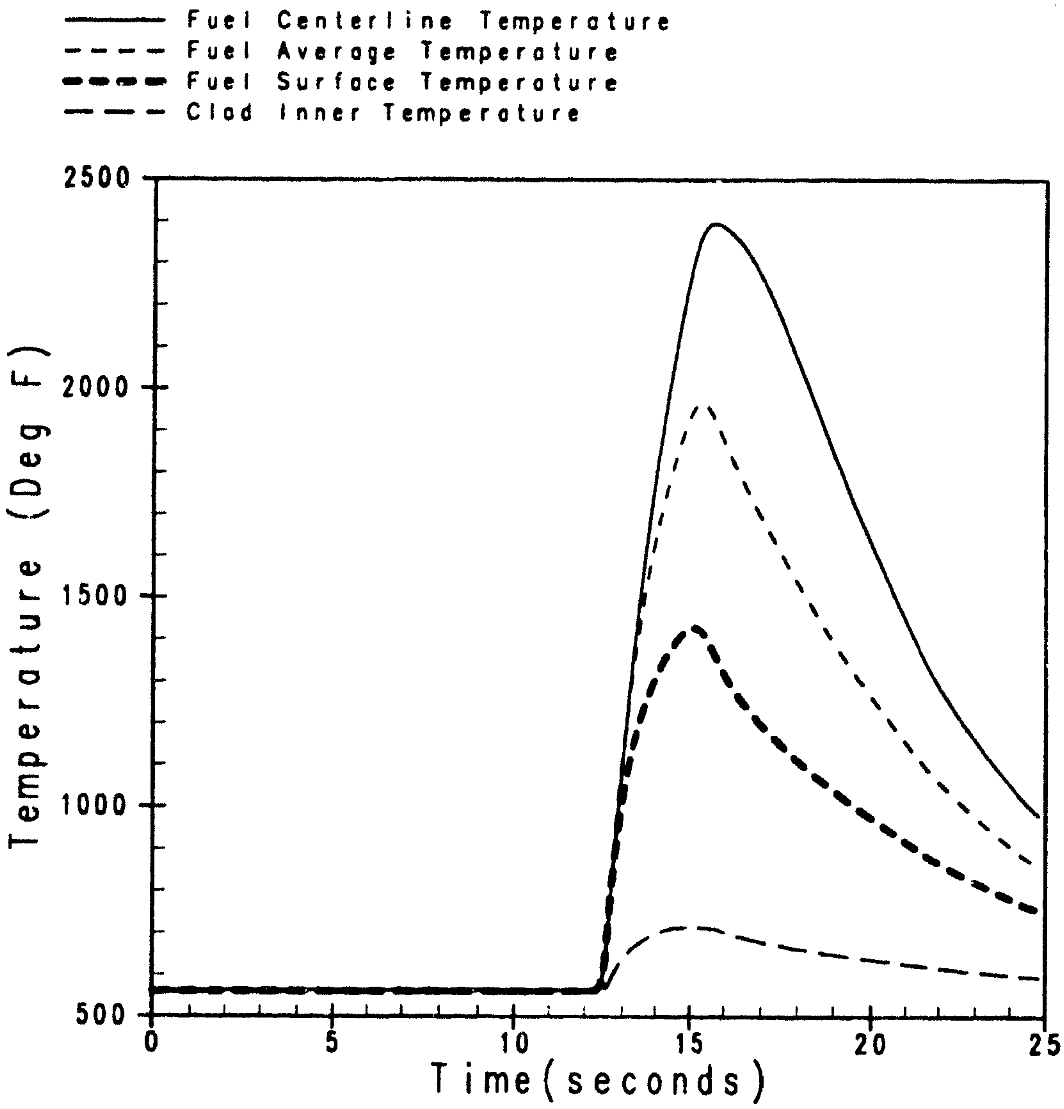


Figure 2.15.2.8.1-3
Fuel Temperature Transients for Uncontrolled Rod Withdrawal
From a Subcritical Condition

2.15.2.8.2 Startup of an Inactive Reactor Coolant Pump (RCP) at an Incorrect Temperature (SRP 15.4.4)

The potential impact of the TPBAR core design on this event is addressed by means of an evaluation. The standard Westinghouse methodology for analysis of the startup of an inactive RCP event is performed using a combination of the LOFTRAN and FACTRAN computer codes, and these results would have only a slight sensitivity to the increase in maximum fuel temperature considered in the current TPBAR core evaluation. No other key safety analysis parameters modeled for this event are affected by the presence of the TPBARs in the core.

The analysis assumptions applied to the startup of an inactive RCP event, as identified in Section 15.4.4 of the reference plant FSAR, are extremely conservative. The Technical Specifications for the reference plant (and all other Westinghouse designed operating plants in the United States) do not permit at power operation with an idle RCP. Despite this, the reference plant analysis for the startup of an inactive RCP event assumes an initial power level of approximately 75% with three out of four RCPs operating. This methodology is consistent with how a four-loop plant might be expected to function if it were licensed for three-loop operation at power, even though no currently operating Westinghouse plants have such a license.

Even with this excessive conservatism, the reference plant analysis results for this event show substantial margin to DNB limits. Given the small magnitude of the fuel temperature increases being considered, the excessive conservatism used in the analysis, and the available margin to DNB limits, the DNB design basis (as defined in Section 4.4 of the reference plant FSAR) will be met for this event with the TPBAR core design.

2.15.2.8.3 Spectrum of Rod Cluster Control Assembly Ejection Accidents (SRP 15.4.8)

Introduction

This accident, as discussed in Section 15.4.8 of the reference plant FSAR, is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a RCCA and drive shaft. The consequences of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. The specific limiting criteria for this event are:

1. Average fuel pellet enthalpy at the hot spot below 200 cal/g for unirradiated and irradiated fuel.

2. Peak reactor coolant pressure less than that which could cause stresses to exceed the faulted condition stress limits.

3. Fuel melting will be limited to less than ten percent (10%) of the fuel volume at the hot spot, even if the average fuel pellet enthalpy is below the limits of Criterion 1 above.

Reactor protection against this event is provided by the power range high neutron flux trip, both high and low settings. A positive flux rate reactor trip function is also available to trip the reactor, but was not simulated.

Method of Analysis

The RCCA Ejection accident for the TPBAR core was reanalyzed following the same basic methodology and using the same computer codes as described in the reference plant FSAR Section 15.4.8. That is, the TWINKLE spatial kinetics computer code is used to predict the core nuclear transient, while the fuel and clad heat transfer is modeled using the FACTRAN computer code. The nuclear power transient predictions of TWINKLE are input to FACTRAN.

The analysis is performed in two stages: First, an average core nuclear power transient calculation, and then a core hot spot calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. The enthalpy and temperature transient in the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is pessimistically assumed to persist throughout the transient. A more detailed discussion of the methodology can be found in the reference plant FSAR.

The key analytical parameters in the analysis are the ejected rod worths and peaking factors, reactivity weighting factors, moderator coefficient, Doppler reactivity, delayed neutron fraction and trip reactivity insertion. All of these parameters are the same as used in the analysis of FSAR Section 15.4.8 of the reference plant FSAR except for the following:

1. A conservatively low value for the Doppler zero to full power reactivity defect. A value of -0.92% Δk is used for the TPBAR core analysis.

2. A normalized curve of trip reactivity versus time is used for current reanalysis which is consistent with the 2.7 seconds drop time associated with the reference plant. The RCCA ejection analysis in the reference plant FSAR conservatively models trip reactivity associated with a longer rod drop time. The total trip reactivity worth modeled remains unchanged from the reference plant FSAR, however.

3. The cases initiated from hot full power conditions include consideration of the slightly higher (maximum increase of 21°F) initial fuel temperatures previously discussed.

The trip reactivity insertion includes the effect of one stuck RCCA adjacent to the ejected rod. The shutdown was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after the high neutron flux trip point was reached. The actual negative reactivity insertion characteristic as a function of time following the trip is consistent with the 2.7 seconds to dashpot entry.

The cases analyzed for the TPBAR core are consistent with those reported in Section 15.4.8 of the reference plant FSAR, and consist of the following:

1. Beginning of Life (BOL), Hot Zero Power
2. BOL, Hot Full Power

3. End of Life (EOL), Hot Zero Power
4. EOL, Hot Full Power

Results

Table 2.15.2.8.3-1 summarizes the results for the cases analyzed. For all the cases, the maximum average fuel pellet enthalpy at the hot spot remains below 200 cal/g. Fuel melt is predicted to occur for both of the cases initiated from hot full power conditions, but the melting is restricted to much less than 10 percent of the pellet volume at the hot spot. The nuclear power and hot spot fuel and clad temperature transients for two cases are presented in Figures 2.15.2.8.3-1 through 2.15.2.8.3-4. The BOL hot full power results are presented since that case produces the maximum fuel stored energy. The EOL hot zero power results are presented simply to be consistent with the information reported in the reference plant FSAR. As shown in Figure 2.15.2.8.3-4, the fuel average temperature actually exceeds the fuel centerline temperature for a short time, early in the transient. This behavior is due to the speed of the transient in conjunction with a power distribution that produces a flux depression at the fuel rod centerline. Neither of the hot zero power cases is limiting and the results for these two cases are very similar. The calculated sequence of events for these cases is presented in Table 2.15.2.8.3-2. For all cases, the post-trip rod insertion occurs after the nuclear power excursion is terminated by Doppler feedback. The reactor remains subcritical following the reactor trip.

The ejection of an RCCA constitutes a break in the RCS, located in the reactor pressure vessel head. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other LOCA to recover from the event.

Conclusions

The results confirm that all the safety analysis criteria for the RCCA ejection event continue to be met for the TPBAR core design. Therefore, the conclusions presented in the reference plant FSAR, for this event, remain valid for the TPBAR core design.

Table 2.15.2.8.3-1 RCCA Ejection Results				
Time in Life→ **Parameter↓**	**BOL-HZP**	**BOL-HFP**	**EOL-HZP**	**EOL-HFP**
Power Level, Fraction of Nominal	0.001	1.02	0.001	1.02
Ejected Rod Worth, % Δk	0.75	0.24	0.84	0.25
β_{eff}, %	0.57	0.57	0.46	0.46
Doppler Feedback Reactivity Weighting	1.744	1.30	3.55	1.30
Trip Reactivity, % Δk	2.0	4.0	2.0	4.0
F_Q before Ejection	—	2.59	—	2.59
F_Q after Ejection	11.0	5.5	26.0	6.0
Number of Operating Pumps	2	4	2	4
Max. Fuel Avg. Temperature, °F	3447	4049	3348	3902
Max. Fuel Center temperature, °F	3970	4968	3803	4866
Max. Fuel Energy, cal/g	145.9	176.9	141.1	169.2
Percent Fuel Melt	0.0	5.27	0.0	4.07

Table 2.15.2.8.3-2 Time Sequence of Events for RCCA Ejection Accident	
Event	**Time (sec.)**
Beginning of Life, full power	
Initiation of rod ejection	0.00
Power range high neutron flux (high setpoint) reached	0.05
Peak nuclear power occurs	0.13
Rods begin to fall into core	0.55
Peak fuel average temperature occurs	2.23
Peak fuel average enthalpy occurs	2.23
Peak clad average temperature occurs	2.33
Peak heat flux occurs	2.34
End of Life, zero power	
Initiation of rod ejection	0.00
Power range high neutron flux (low setpoint) reached	0.20
Peak nuclear power occurs	0.24
Rods begin to fall into core	0.70
Peak clad average temperature occurs	1.69
Peak fuel average temperature occurs	1.90
Peak fuel average enthalpy occurs	1.90
Peak heat flux occurs	1.74

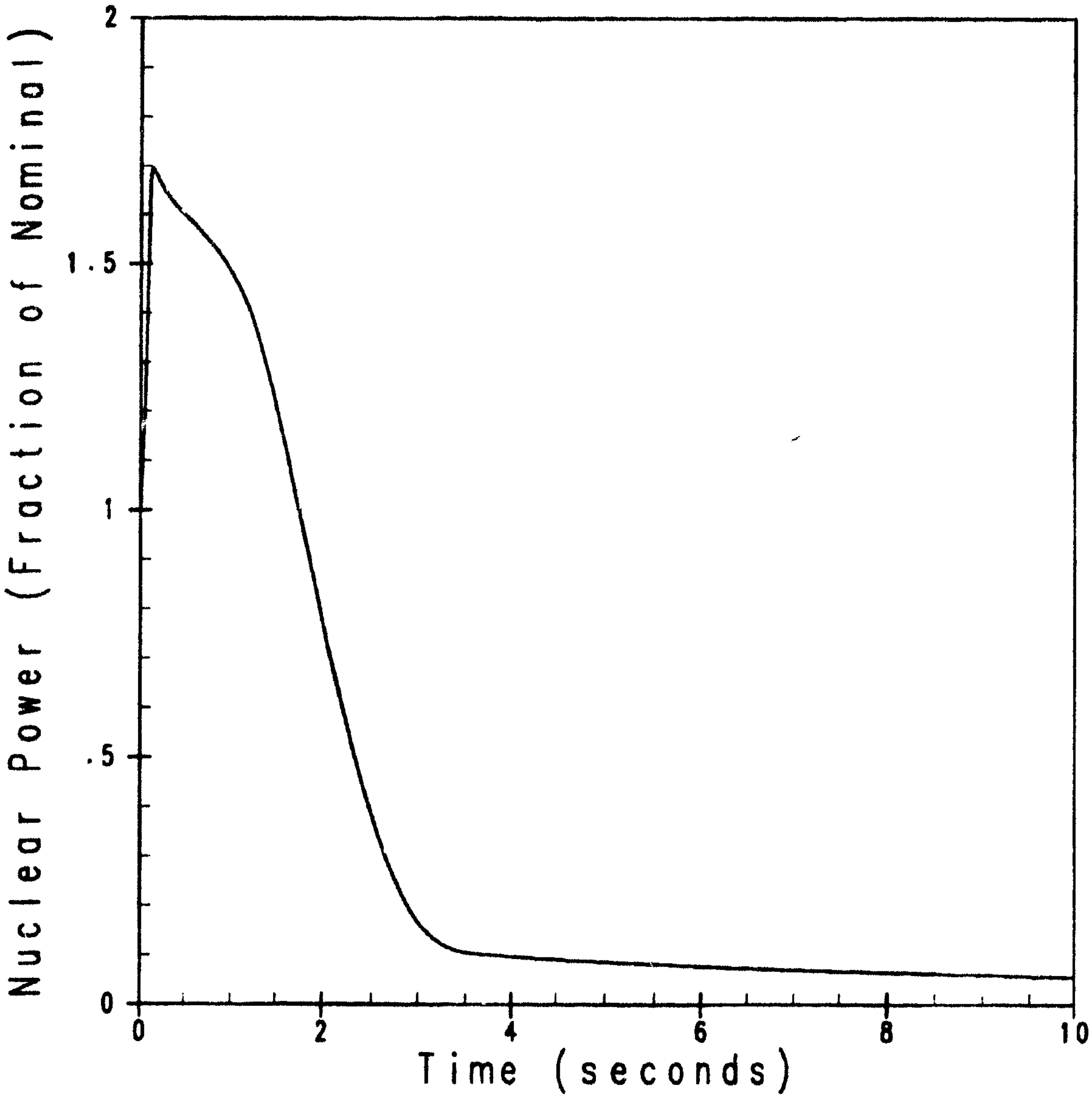


Figure 2.15.2.8.3-1
Nuclear Power Transient for RCCA Ejection
(BOL Full Power)

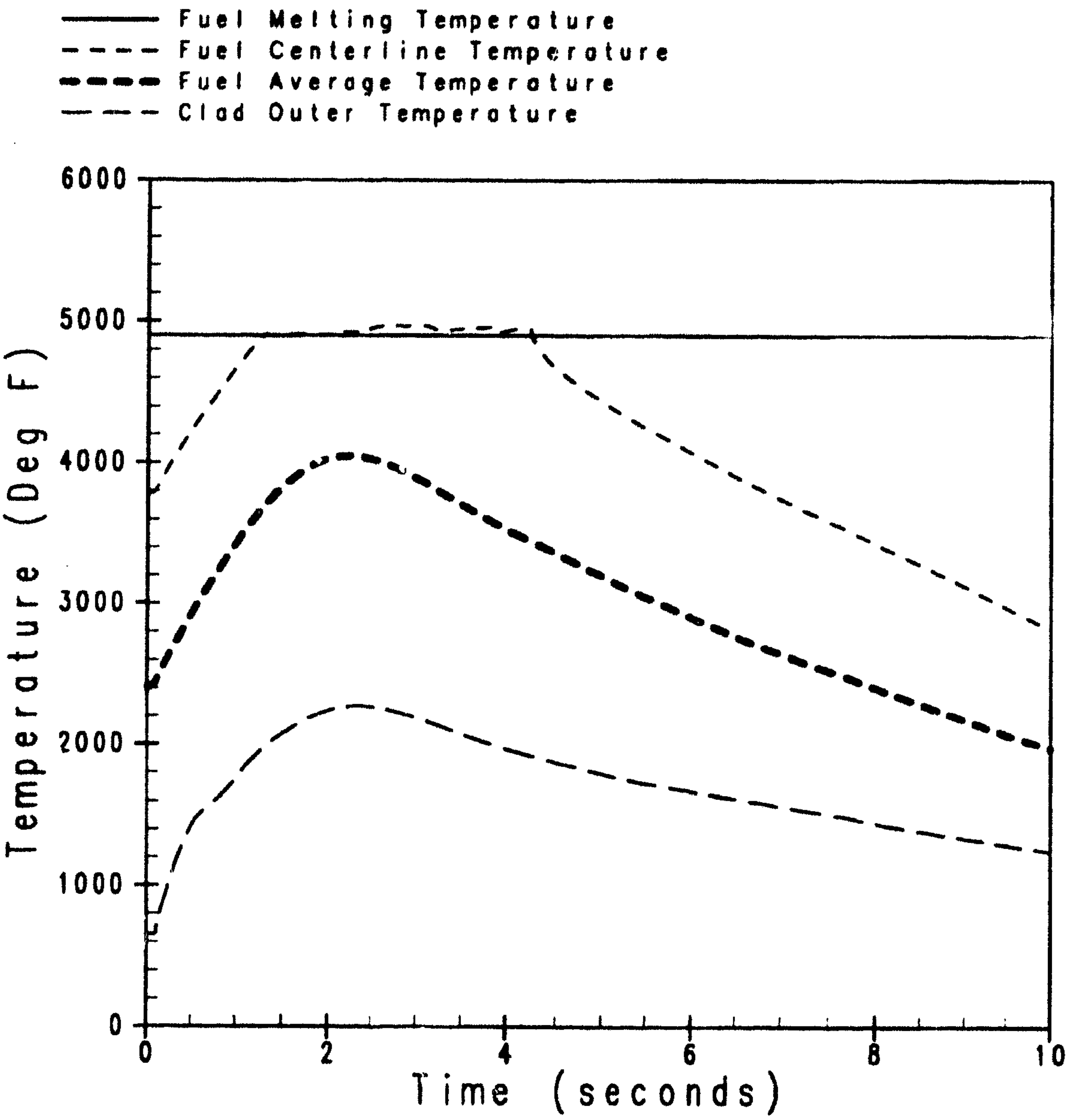


Figure 2.15.2.8.3-2
Fuel Temperature Transients for RCCA Ejection
(BOL Full Power)

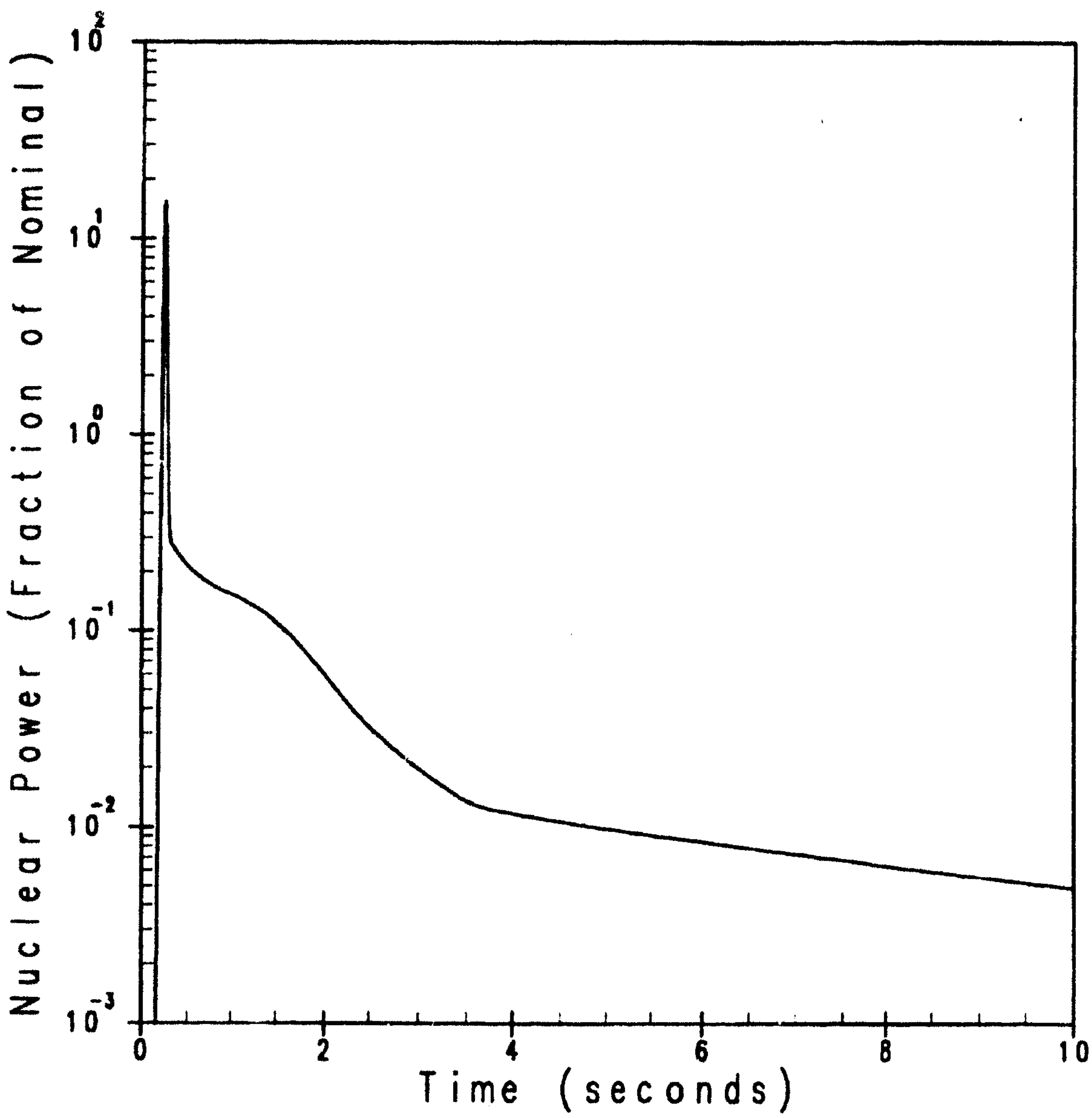


Figure 2.15.2.8.3-3
Nuclear Power Transient for RCCA Ejection
(EOL Zero Power)

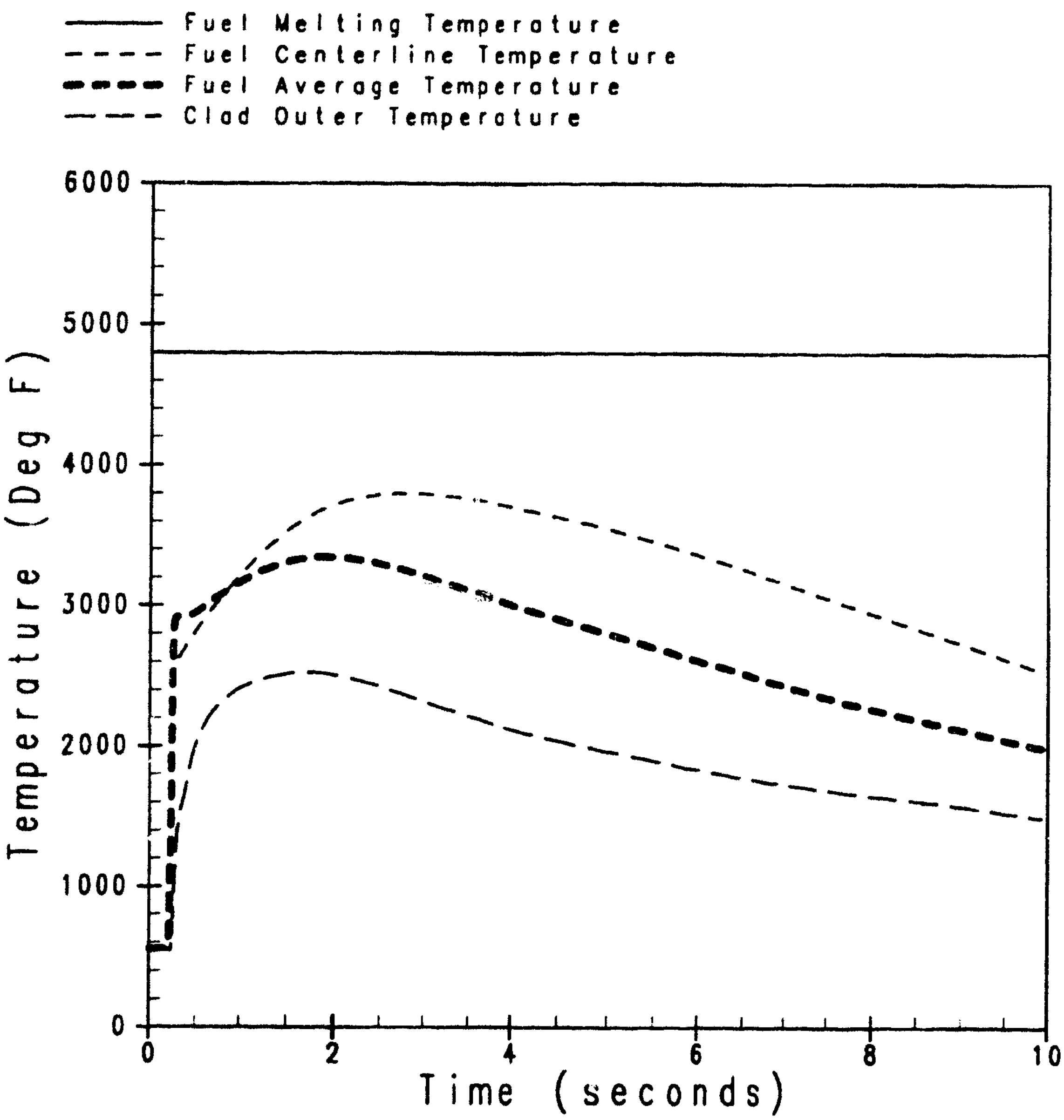


Figure 2.15.2.8.3-4
Fuel Temperature Transients for RCCA Ejection
(EOL Zero Power)

2.15.2.8.4 Steamline Break With Coincident RCCA Withdrawal at Power

The reference plant analysis for this non-LOCA event is documented in Section 15.4.9 of the reference plant FSAR. It should be noted that while this event is included in the reference plant FSAR, the steamline break with coincident RCCA withdrawal at power is not found in the SRP. The analysis for this event was inserted into the reference plant FSAR to address concerns related to non-safety grade equipment being subjected to an adverse environment from high energy line breaks inside or outside containment. This event was postulated as a result of Information Notice IE 79-22 (Reference 11). Individual plants dealt with this event in various ways, but the reference plant chose to incorporate explicit safety analysis into its FSAR. None of the key safety analysis parameters modeled for this event are affected by the presence of the TPBARs in the core. However, the analysis for the steamline break with coincident RCCA withdrawal at power event uses the LOFTRAN computer code and defines a bounding value for the overall fuel to coolant heat transfer coefficient using maximum fuel temperatures. Therefore, the results of the safety analysis will have some sensitivity to the increase in maximum fuel temperature considered in the current TPBAR core evaluation. However, the magnitude of the modeled fuel temperature increase and the resulting change in the overall fuel to coolant heat transfer coefficient are relatively small. A sensitivity study using the reference plant LOFTRAN model has determined that the maximum fuel temperature increase considered in the TPBAR evaluation has an insignificant effect on the predicted results for the steamline break with coincident RCCA withdrawal event. That is, the DNB design basis, as defined in Section 4.4 of the reference plant FSAR, will be met during the steamline break with coincident RCCA withdrawal at power event for the TPBAR core. Therefore, the conclusions in the reference plant FSAR, for this event, remain valid for the TPBAR core design. As indicated above, the treatment of this event is plant specific and, for most plants, the FSAR will not contain direct analysis for a steamline break with coincident RCCA withdrawal at power.

2.15.2.9 Increase In Reactor Coolant Inventory

All of the accidents defined within this event category in the reference plant FSAR are non-LOCA transients that will be evaluated here with regard to impact of the TPBAR core design. The reference plant FSAR does include identification of a group of events, boiling water reactor transients, in FSAR Section 15.5.3, that are not applicable to the reference plant and need not be considered in the current evaluation. For both of the non-LOCA events in the increase in reactor coolant inventory category (listed below) a single evaluation will suffice to address the impact of the TPBAR reload core.

15.5.1 Inadvertent Operation of ECCS During Power Operation

15.5.2 Chemical & Volume Control System Malfunction that Increases Reactor Coolant Inventory

For both these events, the TPBAR core design has not changed any of the bounding values assumed for the key safety analysis parameters used in the reference plant FSAR analyses. Similarly, neither of these events is affected by the increase in maximum fuel temperatures. Therefore, the results for both of these events are be unaffected by the presence of TPBARs in the core.

2.15.2.10 Decrease In Reactor Coolant Inventory

Only a single accident defined within this event category in the reference plant FSAR is a non-LOCA transient that will be evaluated here, with regard to the impact of the TPBAR core design. That event is inadvertent opening of a pressurizer safety or relief valve, and it is documented in Section 15.6.1 of the reference plant FSAR.

For this event, the TPBAR core design has not changed any of the bounding values assumed for the key safety analysis parameters used in the reference plant FSAR analysis. Similarly, the analysis for this event is not affected by the increase in fuel temperatures. Therefore, the results for the inadvertent opening of a pressurizer safety or relief valve event are unaffected by the presence of TPBARs in the core.

2.15.2.11 References

1. WCAP-8330, Westinghouse Anticipated Transients Without Trip Analysis", August 1974.

2. Anderson, T.M., "ATWS Submittal, "Westinghouse Letter NS-TMA-2182 to S.H. Hanauer of the NRC, December 1979.

3. ATWS Final Rule - Code of Federal Regulations 10 CFR 50.62 and Supplementary Information Package, "Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants".

4. Regulatory Guide 1.70 Rev. 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," November 1978.

5. WCAP-9272-P-A (Proprietary) and WCAP-9273-NP-A (Non-proprietary), "Westinghouse Reload Safety Evaluation Methodology", Davidson, S.L. (Ed.), et.al., July 1985.

6. Reference Plant Final Safety Analysis Report Update.

7. WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), Burnett, T. W. T., "LOFTRAN Code Description," April 1984.

8. WCAP 7908-A (Proprietary) and WCAP-7337 (Nonproprietary), "FACTRAN - A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod," Hargrove, H. G., December 1989.

9. WCAP-7979-P-A (Proprietary) and WCAP-8028-A (Non-Proprietary), "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," Risher, D. H., Jr., and Barry, R. F., January 1975.

10. WCAP-11397-P-A (Proprietary) and WCAP-11398-A (Nonproprietary), "Revised Thermal Design Procedure," Friedland, A. J., and Ray, S., April 1989.

11. IE Information Notice 79-22, Qualification of Control Systems", 9/14/79.

2.15.3 INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION (SRP 15.4.7)

2.15.3.1 Acceptance Criteria

This event is classified as a Condition III incident. Condition III events are defined as those events which do not cause more than a small fraction of fuel rods to fail. For the reference plant, the limit on fuel rod failures for Condition III events is 5% of rods (equivalent to about 9.5 fuel assemblies).

2.15.3.2 Discussion

The general discussion of this event for the reference plant, given in Reference 1, applies to the TPC designs as well. As stated in Reference 1, this event comprises core misloading scenarios such as the inadvertent loading of one or more fuel assemblies into improper positions, the loading of a fuel rod during manufacture with one or more pellets of the wrong enrichment, or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment. In addition to these scenarios, misloading events involving burnable absorbers are theoretically possible; for example, the placement of a cluster of 20 burnable absorbers into a core location slated to have 24 burnable absorbers. Misloading errors can lead to increased heat fluxes at the location of the misloading if the misloading results in a local reactivity increase relative to the intended pattern. If the misloading results in a local reactivity decrease, heat flux increases away from the location of the misloading are possible due to unintended power tilts. These kinds of increases, however, are generally distributed and small relative to those where the local reactivity is increased.

To reduce the probability of core loading errors, each fuel assembly and core component is marked with an identification number and loaded in accordance with a core loading diagram. During core loading, the identification numbers are checked before each assembly is moved into the core. Identification numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placement after the loading is completed. These procedures make the likelihood of core misloadings very small.

Should misloadings occur, however, the incore system of movable flux detectors, which is used to verify power shapes during startup and throughout the cycle, is capable of revealing enrichment errors or misloadings which would cause the kind of substantial power distribution perturbation that would be necessary to induce significant fuel rod failures. In addition, thermocouples placed at the core outlet provide an additional indication of power distribution anomalies. This incore instrumentation along with the startup testing performed each cycle make the detection of significant misloadings highly likely.

2.15.3.3 Evaluation

For the TPC cores, a spectrum of misloading scenarios were examined that are similar to those in Reference 1. Figures 2.15.3-1 through 2.15.3-6 show the results of six misloading scenarios for the TPC first cycle (see Figure 2.4.3-2 in Section 2.4.3 for the reference loading pattern). The scenarios are as follows:

1. interchange two feed assemblies in interior locations (MED2 and MED1 assemblies were interchanged.)

2. interchange two feed assemblies near the core periphery (HIGH and MED2 assemblies were interchanged.)

3. misload 20 and 24 TPBAR clusters near the core periphery (clusters exchanged)

4. misload 20 and 24 TPBAR clusters in the core interior (clusters exchanged)

5. enrichment error in the core interior (MED2 assembly used in place of MED1 assembly)

6. enrichment error near the core periphery (MED1 assembly used in place of a MED2 assembly)

The figures show the HZP assembly power percent differences in instrumented locations relative to the power distribution for the correctly loaded core.

As Figure 2.15.3-1 shows, the feed assemblies interchanged in Case 1 were comparable enough in terms of reactivity that only a very small power distribution perturbation resulted. In Case 2, the misloaded feed assemblies caused a detectable power distribution perturbation near the core periphery. In Case 3, the swapping of 20 and 24 TPBAR clusters has only a small local power distribution effect near the core periphery. In Case 4, a larger but still relatively small local power distribution increase was observed in the interior of the core where a 24 TPBAR cluster was replaced with a cluster of 20. The magnitude of this perturbation would not cause design limits to be exceeded. In both Cases 5 and 6, the enrichment errors caused power distributions perturbations which were small but likely significant enough to be detected due to the asymmetry in the core power distribution. It should be noted that the power distribution perturbations shown in the figures would be even smaller at HFP conditions due to feedback effects.

A similar set of scenarios were examined for the TPC equilibrium cycle. Figures 2.15.3-7 through 2.15.3-12 show the results (see Figure 2.4.3-4 in Section 2.4.3 for the reference loading pattern). The cases are as follows:

1. interchange two once-burned assemblies in interior locations

2. interchange two feed assemblies in the core interior (H0X and L0X assemblies were interchanged.)

3. interchange two feed assemblies near the core periphery (H0X and L0X assemblies were interchanged.)

4. misload 20 and 24 TPBAR clusters near the core periphery (clusters exchanged)

5. misload 20 and 24 TPBAR clusters in the core interior (clusters exchanged)

6. enrichment error near the core periphery (high enrichment feed assembly used in place of a low enrichment feed assembly)

As in the first cycle scenarios, some of these equilibrium cycle scenarios produced only very small power distribution perturbations; others produced perturbations which would likely be detectable. Cases 1 and 2 for the equilibrium cycle (see Figures 2.15.3-7 and 2.15.3-8) caused only m changes in the power distribution. Case 3 produced a significant power dist change near the core periphery but still within design limits. Cases 4 and 5, in wh R clusters were interchanged, produced small, local power distribution changes wh not cause design limits to be exceeded. Case 6, the enrichment error case, produced a significant power increase near the core periphery which would likely be detected.

Fuel assembly enrichment errors would be prevented by administrative procedures and product inspection implemented in fabrication. While the above scenarios did not examine the effect of a single pin or fuel pellet with a higher enrichment than the nominal value, the consequences of this kind of undetected enrichment error in terms of reduced DNBR and increased fuel and clad temperatures would be limited to the incorrectly loaded pin or pins.

2.15.3.4 Conclusion

Fuel assembly and burnable absorber loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error should occur, power distribution perturbations large enough to challenge fuel limits and cause significant fuel failures would be readily detectable by incore instrumentation during startup testing. Continued operation with a perturbation of a detectable magnitude would be evaluated.

2.15.3.5 Reference

1. Reference Plant, Final Safety Analysis Report.

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.0			0.1					
2			0.1			0.1		0.0							
3								0.0		0.0		0.0		0.1	
4		0.0	0.0					0.0							
5					-0.1				-0.1		0.0		0.0		
6	0.1		0.0			0.0		0.0						0.1	
7				-0.1			0.0			0.1			0.0		
8 (90°)	0.0		-0.1		-0.1		-0.3			0.1		0.0	0.0	0.0	
9		0.0							0.5		0.0				0.0
10					-0.1		-0.3					0.0			
11	0.1				-0.1			0.0			0.0				0.1
12						-0.1			0.0			0.0			
13			0.0		0.0			0.0						0.1	
14			0.1				0.0			0.1		0.0			
15					0.1			0.1							

0°

Note: Only movable detector locations shown. The numbers represent the percent deviation in the assembly average power. Shaded assemblies indicate the location of misloading.

Figure 2.15.3-1
First Cycle Misloading Case 1
Interchange Feed Assemblies Inboard

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-0.8			-0.8					
2			-0.9			-0.9		-0.8							
3								-0.8		-0.7		-0.5		-0.3	
4		-1.1	-1.0					-0.8							
5					-1.1				-0.6		-0.3		0.0		
6	-1.2		-1.2			-1.1		-0.7						0.6	
7				-1.2			-0.9			0.1			1.1		
8 (90°)	-1.3		-1.4		-1.3		-0.9			0.6		1.5	1.9	2.1	
9	.	-1.4							0.3		1.8				3.4
10					-1.5		-0.9					3.7			
11	-1.6				-1.6			-0.5			3.8				5.9
12						-1.6			-0.1			9.3			
13			-1.9		-1.9			-1.6						11.2	
14			-1.9				-2.0			-4.2		6.8			
15					-2.0			-2.6							

0°

Note: Only movable detector locations shown. The numbers represent the percent deviation in the assembly average power. Shaded assemblies indicate the location of misloading.

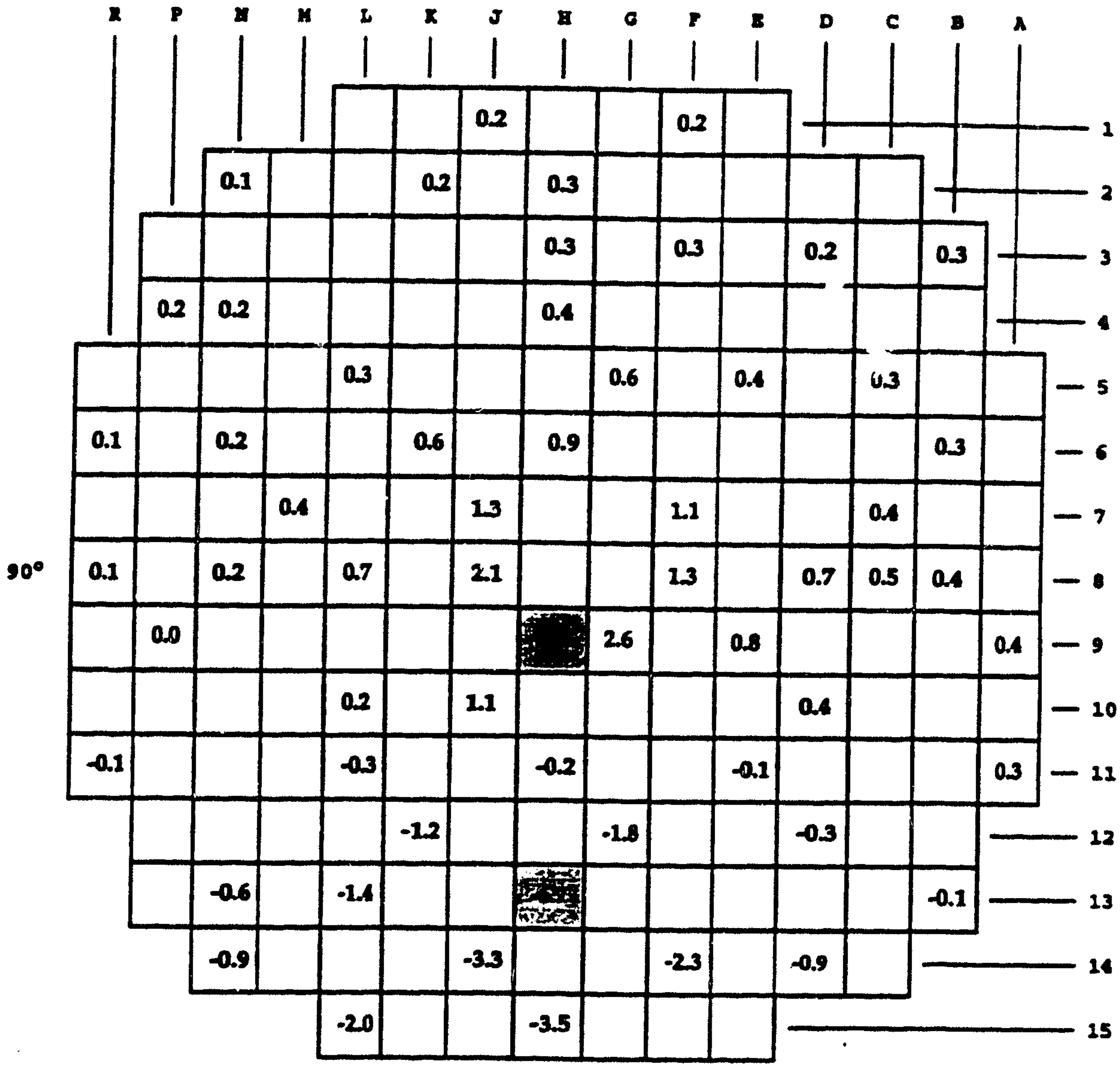
Figure 2.15.3-2
First Cycle Misloading Case 2
Interchange Feed Assemblies Outboard

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.2			0.1					
2			0.1			0.1		0.1							
3								0.0		0.1		0.1		0.1	
4		0.0	0.1					0.0							
5					0.0				-0.1		0.0		0.0		
6	0.1		0.0			-0.1		-0.1						0.1	
7				-0.1			-0.2			-0.1			0.0		
8 (90°)	0.1		-0.1		-0.2		-0.2			-0.2		0.0	0.1	0.1	
9		0.0							-0.3		-0.1				0.2
10					-0.4		-0.6					0.1			
11	-0.1				-0.5			-0.7			0.2				0.3
12						-0.9			-0.4			0.6			
13			-0.4		-0.8									0.7	
14			-0.5				-0.8			2.7		1.4			
15					-0.6			1.3							

0°

Note: Only movable detector locations shown. The numbers represent the percent deviation in the assembly average power. Shaded assemblies indicate the location of misloading.

Figure 2.15.3-3
First Cycle Misloading Case 3
Misload 20 and 24 TPBAR Clusters Outboard



Note: Only movable detector locations shown. The numbers represent the percent deviation in the assembly average power. Shaded assemblies indicate the location of misloading.

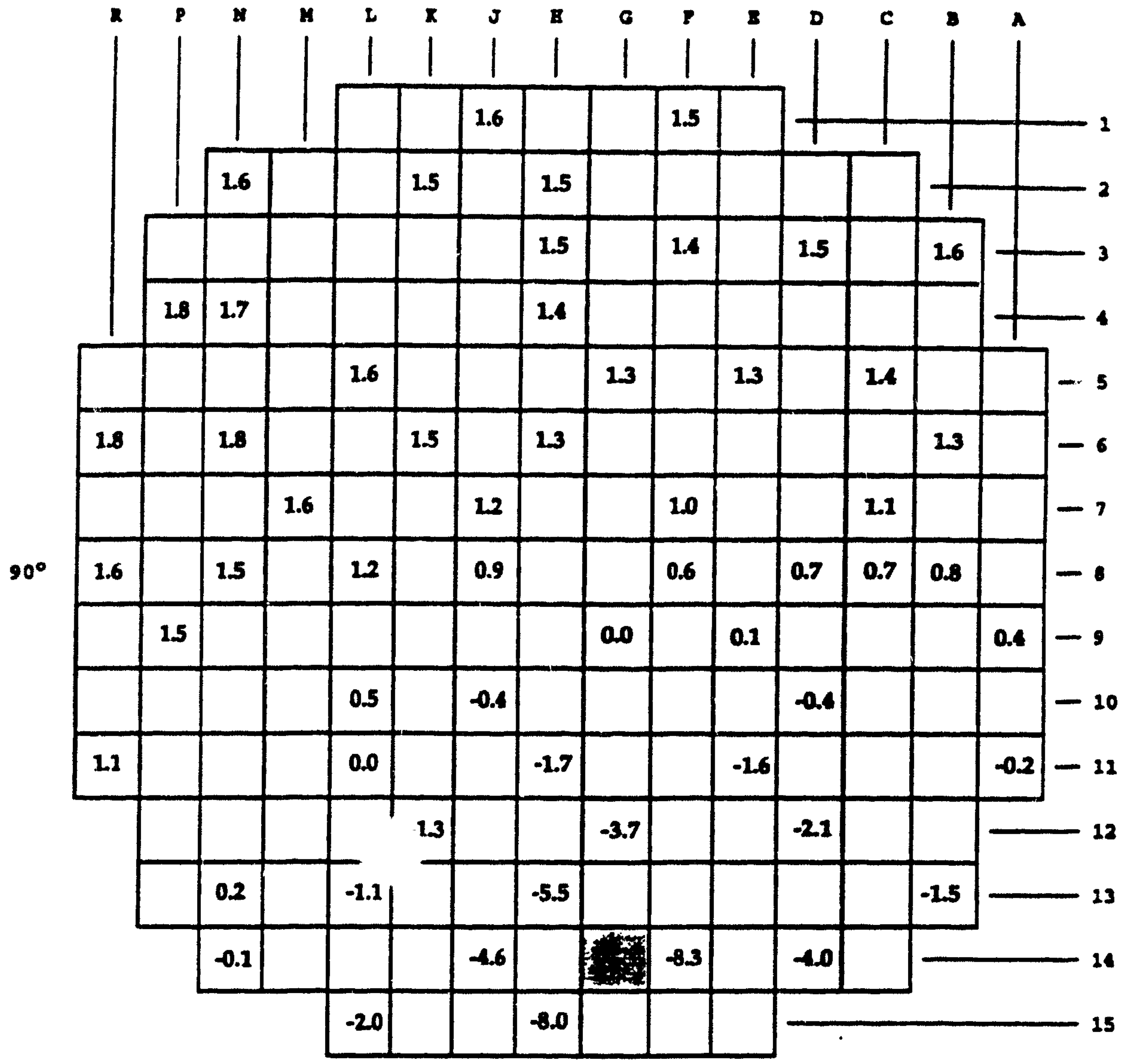
Figure 2.15.3-4
First Cycle Misloading Case 4
Misload 20 and 24 TPBAR Clusters Inboard

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-2.5			-2.2					
2			-2.9			-2.7		-2.4							
3								-2.4		-2.1		-1.7		-1.5	
4		-2.8	-2.9					-2.1							
5					-2.4				-1.8		-1.6		-1.4		
6	-2.5		-2.4			-1.9		-1.6						-1.3	
7				-1.8			-1.0			-0.9			-1.2		
8 (90°)	-1.9		-1.5		-0.7		0.0			-0.4		-0.8	-1.0	-1.1	
9		-1.2							0.8		-0.1				-1.0
10					1.6		3.3					-0.2			
11	-0.4				3.0			5.0			0.9				-0.6
12						6.7			3.8			0.6			
13			2.0		4.4			6.1						0.1	
14			2.6				5.7			2.7		1.1			
15					4.7			4.7							

0°

Note: Only movable detector locations shown. The numbers represent the percent deviation in the assembly average power. Shaded assembly indicates the location of enrichment error.

Figure 2.15.3-5
First Cycle Misloading, Case 5
Enrichment Error Inboard



Note: Only movable detector locations shown. The numbers represent the percent deviation in the assembly average power. Shaded assembly indicates the location of enrichment error.

Figure 2.15.3-6
First Cycle Misloading Case 6
Enrichment Error Outboard

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.1			0.1					
2			0.0			0.1		0.0							
3								0.1		0.1		0.0		0.0	
4		0.0	0.0					0.3							
5					0.2				0.6		0.2		0.1		
6	0.1		0.1			0.6		1.3						0.1	
7				0.2			1.7			0.9			0.1		
8 (90°)	0.1		0.1		0.3		1.9			0.9		0.1	0.1	0.0	
9		0.0							0.3		0.0				0.0
10					-0.3		-1.6					-0.2			
11	0.0				-0.4			-2.6			-0.5				0.1
12						-0.7			-1.2			-0.3			
13			-0.1		-0.3			-0.7						-0.2	
14			-0.2				-0.4			-0.3		-0.1			
15					-0.1			-0.3							

0°

Note: Only movable detector locations shown. The numbers represent the percent deviation in the assembly average power. Shaded assemblies indicate the location of misloading.

Figure 2.15.3-7
Equilibrium Cycle Misloading Case 1
Interchange Burned Assemblies Inboard

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.5			0.6					
2			0.5			0.5		0.5							
3								0.5		0.5		0.5		0.5	
4		0.4	0.4					0.5							
5					0.3				0.5		0.5		0.5		
6	0.2		0.3			0.2		0.5						0.4	
7				0.0			0.2			0.6			0.4		
8 (90°)	0.1		-0.1		-0.4		-0.6			0.6		0.3	0.3	0.3	
9		-0.1							1.4		0.3				0.1
10					-0.8		-1.0					-0.1			
11	-0.2				-0.8			-0.3			-0.3				-0.1
12						-0.9			-0.5			-0.5			
13			-0.8		-0.9			-0.7						-0.7	
14			-0.9				-0.8			-0.6		-0.8			
15					-0.9			-0.7							

0°

Note: Only movable detector locations shown. The numbers represent the percent deviation in the assembly average power. Shaded assemblies indicate the location of misloading.

Figure 2.15.3-8
Equilibrium Cycle Misloading Case 2
Interchange Feed Assemblies Inboard

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-1.9			-1.9					
2			-2.0			-1.9		-1.9							
3								-1.8		-1.8		-1.7		-1.5	
4		-2.1	-2.0					-1.8							
5					-1.9				-1.5		-1.3		-1.2		
6	-2.0		-1.9			-1.6		-1.3						-0.8	
7				-1.8			-1.2			-0.6			-0.4		
8	-2.0		-1.8		-1.5		-0.9			0.0		0.2	0.3	0.2	
9		-1.8							0.4		1.0				0.8
10					-1.1		0.0					2.3			
11	-1.7				-0.8			1.2			4.2				2.0
12						0.0			3.2			8.2			
13			-1.2		-0.3			2.4						0.3	
14			-0.9				1.6			6.7		12.6			
15					0.1			2.9							

90°

0°

Note: Only movable detector locations shown. The numbers represent the percent deviation in the assembly average power. Shaded assemblies indicate the location of misloading.

Figure 2.15.3-9
Equilibrium Cycle Misloading Case 3
Interchange Feed Assemblies Outboard

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.0			0.2					
2			-0.3			-0.1		0.0							
3								0.1		0.3		0.3		0.5	
4		-0.4	-0.3					0.0							
5					-0.4				0.2		0.4		0.5		
6	-0.5		-0.5			-0.2		0.1						0.7	
7				-0.6			0.0			0.5			0.8		
8	-0.7		-0.7		-0.6		-0.1			0.5		0.8	0.9	1.0	
9		-0.9							0.3		0.8				1.1
10					-1.0		-0.6					1.3			
11	-1.0				-1.3			-0.8			1.6				1.6
12						-1.9			0.0			2.3			
13			-1.6		-1.9									2.1	
14			-1.7				-2.6			2.8		3.0			
15					-2.1			-1.7							

90°

0°

Note: Only movable detector locations shown. The numbers represent the percent deviation in the assembly average power. Shaded assemblies indicate the location of misloading.

Figure 2.15.3-10
Equilibrium Cycle Misloading Case 4
Interchange 20 and 24 TPBAR Cluster Outboard

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.5			0.4					
2			0.2			0.4		0.4							
3								0.4		0.3		0.3		0.2	
4		0.2	0.3					0.5							
5					0.4				0.7		0.5		0.3		
6	0.2		0.3			0.8		1.0						0.3	
7				0.4			1.5			1.0			0.3		
8 (90°)	0.2		0.3		0.8		2.2			1.3		0.4	0.3	0.2	
9		0.2							2.6		0.7				0.1
10					0.3		1.3					0.1			
11	0.0				-0.3			-0.4			-0.3				0.1
12						-1.4			-2.2			-0.6			
13			-0.7		-1.4									-0.4	
14			-0.9				-3.7			-2.6		-1.2			
15					-2.1			-3.9							

0°

Note: Only movable detector locations shown. The numbers represent the percent deviation in the assembly average power. Shaded assemblies indicate the location of misloading.

Figure 2.15.3-11
Equilibrium Cycle Misloading Case 5
Interchange 20 and 24 TPBAR Cluster Inboard

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-2.0			-1.9					
2			-2.2			-2.2		-2.0							
3								-2.0		-1.9		-1.6		-1.5	
4		-2.2	-2.2					-1.8							
5					-2.1				-1.5		-1.3		-1.1		
6	-2.0		-2.1			-1.7		-1.3						-0.6	
7				-1.9			-1.2			-0.4			0.0		
8 (90°)	-1.8		-1.8		-1.4		-0.8			0.1		0.6	0.7	0.8	
9		-1.7							0.4		1.2				1.9
10					-0.9		0.0					2.8			
11	-1.4				-0.6			1.1			3.5				4.5
12						0.3			2.2			6.6			
13			-0.8		0.0			1.8						11.2	
14			-0.4				1.5			3.9		8.4			
15					0.6			2.3							

0°

Note: Only movable detector locations shown. The numbers represent the percent deviation in the assembly average power. Shaded assemblies indicate the location of enrichment error.

Figure 2.15.3-12
Equilibrium Cycle Misloading Case 6
Enrichment Error Outboard

2.15.4 STEAM GENERATOR TUBE FAILURE (SRP 15.6.3) EVALUATION

The licensing basis steam generator tube rupture (SGTR) analysis for the reference plant is performed to demonstrate that margin to steam generator overfill exists and that the radiological consequences are within the guideline values. Input into the SGTR analysis includes initial plant operating conditions, reactor core physics characteristics, and reactor control and protection system setpoints and uncertainties. The reactor core characteristics have only a minor impact on the SGTR analysis and, in addition, the core characteristics which have an impact on the SGTR are not significantly affected by the incorporation of a full complement of TPBARs. Since there are no changes to the operating conditions or setpoints, the SGTR thermal and hydraulic analysis currently in effect for the reference plant will continue to apply to the TPC. The radiological consequences of the SGTR are discussed in Section 2.15.6.4.

2.15.5 LOCA (SRP 15.6.5) EVALUATIONS

2.15.5.1 TPBAR Response to Large and Small Break LOCA

2.15.5.1.1 Introduction

In order to assess the potential for interaction of the TPBARs with the LOCA transients, it was necessary to first estimate the response of the TPBARs to the design basis LOCAs, both large and small breaks. The TPBAR generates minimal heat during a LOCA and is heated primarily by radiation from the fuel rods to the fuel assembly guide thimble and radiation from the thimble across the gap to the TPBAR. Convection of the steam and entrained liquid, on the outer thimble surface, provides cooling comparable to that experienced by the fuel rods. To provide estimates of these effects, a fuel rod response code (LOCTAJR) was modified to include the fuel assembly thimble tube and the TPBAR in the calculations. Fuel rod temperatures and fluid conditions were used as boundary conditions and were obtained from the standard Appendix K LOCA analyses. These analyses were performed using the '81 Evaluation Model plus BASH for large break (SATAN, BASH, and LOCBART Codes, Refs. 6, 7, 8, and 9) and the Westinghouse Small Break Evaluation Model (NOTRUMP and SBLOCTA Codes, Refs. 10, 11, and 9) for small break. The heatup of the TPBAR was modeled in a conservative fashion using assumptions consistent with Appendix K methodology and selected to maximize the TPBAR thermal response.

2.15.5.1.2 Methodology

Calculation of the TPBAR temperatures is performed with a modified fuel rod response code, LOCTAJR, which uses as boundary conditions the cladding temperature of the surrounding fuel rods and the core steam and entrained liquid convective heat transfer coefficients and temperatures. The boundary conditions are taken from the Appendix K LOCA analyses for the reference plant. In the modified LOCTAJR, the fuel assembly thimble and TPBAR are modeled.

The following modeling assumptions are made due to the component geometry and the pertinent heat transfer mechanisms:

1. Steam flow in the annulus between the TPBAR and the thimble will be minimal due to (1) the low heat generation rate in the TPBAR and resulting low steaming rates in the annulus and (2) the tendency of TPBAR swelling to block the annulus during LOCA conditions. Since steam

flow in the annulus would tend to reduce the TPBAR temperatures, it is conservatively neglected.

2. Temperature calculations in the thimble and TPBAR can be performed 1-dimensionally at the elevations of high fuel rod temperature since axial conduction effects are negligible.

3. Heat transfer to the outer surface of the thimble will include radiant heat transfer from the fuel rods and convective cooling from the core steam and entrained liquid flows. The fuel rod temperatures and fluid conditions are boundary conditions to the calculations and are obtained from the Appendix K LOCA analyses.

4. Heat transfer in thimble/TPBAR annulus consists of radiation and conduction through the steam.

5. Zirconium/water oxidation will be calculated on the exterior surface of the thimble. In the thimble/ TPBAR annulus, oxidation of the thimble will be neglected due to the lack of significant steam flow.

6. Heat generation in the TPBAR is included in the thermal calculations although the post-LOCA heating rates in the TPBAR are negligible.

7. Due to the high thermal conductivity of gases within the TPBAR and the low heatup rates, temperature gradients inside the TPBAR are minimal. The mean heat capacity of the TPBAR is input as the product of layer weighted density and specific heat, and a mean temperature is calculated.

2.15.5.1.3 Analysis Inputs and Results

The calculations for large and small break LOCAs are based on the fuel temperatures from Appendix K LOCA analyses. The large break analysis is based upon the current analysis of record for the reference plant. The small break analysis was performed with an analysis which included an updated NRC approved evaluation model containing the COSI Steam Condensation Model and modeling safety injection (SI) in the broken loop.

In the large break loss of coolant accident (LBLOCA) analysis of the fuel, performed according to the requirements of Appendix K of 10 CFR 50, the Peak Cladding Temperature (PCT) reached 1937°F. The hot spot on the TPBARs in the hot assembly were calculated to reach 1833°F as shown in Figure 2.15.5-1. As a consequence, bursting of the TPBAR cladding would be expected at approximately 120 seconds into the transient. In the TPBAR analysis, it was assumed that swelling would close the annular gap between the thimble and TPBAR, precluding steam flow in the gap. Thus, zero steam flow was assumed in the analysis and no benefit obtained for direct steam cooling of the TPBARs. This provides a consistent and conservative treatment of the effects of TPBAR burst and blockage on the TPBAR temperatures.

In the small break loss of coolant accident (SBLOCA) analysis of the fuel, performed according to the requirements of Appendix K of 10 CFR 50, the Peak Cladding Temperature (PCT) reached 1465°F. The hot spot on the TPBARs in the hot assembly was calculated to reach 1447°F as shown in Figure 2.15.5-2. During the period when the TPBAR cladding temperature exceeds 650°F, the

minimum system pressure is 560 psia (see Figure 2.15.5-3). The minimum pressure during the transient is 550 psia and occurs subsequent to 3500 seconds. As a consequence, bursting of the TPBAR cladding would not be expected. The reactor vessel mixture level, shown in Figure 2.15.5-4, shows a minimum core level of approximately 8 feet prior to the time of peak cladding temperature at 1800 seconds.

Some swelling of the TPBAR cladding might occur in the vicinity of the hot spot, however blockage of the thimble annulus is uncertain. In the analysis, it was assumed that this swelling would close the annular gap between the thimble and TPBAR, precluding steam flow in the gap. This provides a conservative treatment of the effects of TPBAR burst and blockage on the TPBAR temperatures since including steam flow effects in the annulus would provide a slight reduction of the TPBAR temperatures.

2.15.5.2 Interaction of TPBARs with Large Break LOCA

The introduction of TPBARs into a standard plant has the potential for impacting the LBLOCA analysis which demonstrates compliance with the LOCA criteria defined in 10 CFR 50.46(b). These criteria include the following items:

1. The Peak Cladding Temperature (PCT) shall not exceed 2200°F.

2. The maximum local cladding oxidation shall nowhere exceed 17% of initial cladding thickness.

3. The total cladding oxidation shall not exceed 1% of the fuel cladding surrounding the active fuel.

4. Changes in core geometry shall be such that the core remains amenable to cooling.

5. Following actuation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value.

The TPBAR core has the potential for impacting the standard LBLOCA analysis through the following effects:

a. The TPBARs have been observed to have a slight effect on the core axial power distributions. Bounding axial power distributions, based on standard core designs, are used in LBLOCA calculations which demonstrate compliance with 10 CFR 50.46(b)(1) through (3). These standard LBLOCA core axial power distributions must bound the TPBAR core axial power distributions.

b. The core radial power distribution, or radial power census, is required for use in the core wide ZrO_2 calculation which must be less than 1%, per 10 CFR 50.46(b)(3), and is potentially affected by the TPBAR core design.

c. The swelling and burst effects of the TPBARs within the thimbles and the adjacent fuel channels has the potential to affect core blockage and coolable geometry considerations. Coolable geometry must be maintained as required by 10 CFR 50.46(b)(4).

The Long Term Cooling LOCA evaluation, required by 10 CFR 50.46(b)(5), is not impacted by the application of TPBARs.

Axial Power Distribution Effects of the TPBARs

As indicated in Section 2.4.3, the use of TPBARs tends to produce some reduction in axial peaking and flattening of the axial power profile as compared to a standard core. This is most apparent at both BOL and EOL for the equilibrium cycle. In addition, there appears to be less peaking at the top of the core, a condition which is also beneficial in LOCA analysis. A comparison of the TPBAR core axial power shapes with those used for analysis of standard cores, indicates that the TPBAR shapes are bounded by those used for standard plant LBLOCA analysis. The standard core shapes were selected to be limiting with respect to LOCA considerations and then scaled to the Technical Specifications limits as required by Appendix K of 10 CFR 50.

Radial Power Distribution Effects of the TPBARs

The introduction of TPBARs into the core has the potential for impacting the core radial rod power census, which is an input to the post-LOCA core wide ZrO_2 calculation. This calculation is required to show compliance with the 10 CFR 50.46(b)(3) limit of < 1% of all zirconium metal surrounding the fuel. A conservative core radial rod power census based on non-TPBAR core designs is an input to the post-LOCA core wide ZrO_2 calculation. This radial power census is applied as a factor to the ZrO_2 calculations for the hot rod. With the introduction of TPBARs in all but control rod locations, the potential for flattening of the core radial power distribution exists. The methodology used for the post-LOCA core wide ZrO_2 calculation was reviewed to ensure that the TPBAR core was conservatively addressed. The standard methodology was found to be sufficiently conservative and the core wide ZrO_2 reaction to be well less than 1% for the reference plant.

Swelling and Burst Effects of the TPBARs

Sensitive information has been removed

The swelling and burst behavior of the stainless steel clad TPBARs, over the anticipated range of internal pressures, is expected to be similar to that observed for Zr-4 (References 1 through 3) and stainless clad fuel rods. Since the flow channels adjacent to the thimbles are formed by three fuel rods and one thimble, the effects of fuel rod swelling and burst will have the major influence on total channel blockage and the effects of TPBAR and the resultant thimble swelling will be less significant. Blockage of the channels adjacent to the thimble should be equal or less than predicted for a typical fuel rod channel and coolable geometry is not a concern. However, it is noted that some fragmentation of the TPBAR cladding may occur. In spite of the shielding provided by the thimble tubing, some particle impact on the adjacent fuel is possible. This should be insignificant since burst of the fuel rod cladding in the hot assembly, during the LBLOCA, occurs at approximately 60 seconds and all hot assembly rods will have burst prior to TPBAR cladding failure. Thus, any particle impacts on the adjacent fuel rods would have no significant effect. (It is noted that fuel rod burst elevation and TPBAR burst elevation would differ due to the variation in burst time and the shift in the hot spot to higher core elevations with time.)

2.15.5.3 Interaction of TPBARs with Small Break LOCA

The TPBAR core has the potential for impacting the standard SBLOCA analysis, similarly to the LBLOCA, through the following effects:

a. The TPBARs have been observed to have a slight effect on the core axial power distributions. Bounding axial power distributions, based on standard core designs, are used in SBLOCA calculations which demonstrate compliance with 10 CFR50.46(b)(1) through (3). These standard SBLOCA power axial power distributions must bound the TPBAR core power distributions.

b. The core radial power distribution, or radial power census, is required for use in the core wide ZrO_2 calculation which must be less than 1%, per 10 CFR 50.46(b)(3), and is potentially affected by the TPBAR core design.

c. The swelling and burst effects of the TPBARs within the thimbles and the adjacent fuel channels has the potential to affect core blockage and coolable geometry considerations. Coolable geometry must be maintained as required by 10 CFR 50.46(b)(4).

The Long Term Cooling LOCA evaluation, required by 10 CFR 50.46(b)(5), is not impacted by the application of TPBARs.

The calculation of the core wide ZrO_2 is based on the same methodology for both large and small break LOCAs. The acceptability of the standard methodology for large breaks applies equally to small breaks.

<u>Axial Power Distribution Effects of the TPBARs</u>

In contrast to the limiting LBLOCA axial power shapes, the limiting small break power shape always peaks near the top of the core since core uncovery is typically limited to no more than the top few feet. Also, with the Westinghouse SBLOCA Evaluation Model, fuel assembly grid effects

are not significant in small break analyses. Thus, the methodology for selecting the limiting axial power shapes differs somewhat between the large and small break evaluation models.

As indicated in Section 2.4.3, the use of TPBARs tends to produce some reduction in axial peaking and flattening of the axial power profile as compared to a standard core. This is most apparent at both BOL and EOL for the equilibrium cycle. In addition, there appears to be less peaking at the top of the core, a condition which is highly beneficial in SBLOCA analysis. A comparison of the TPBAR core axial power shapes with those used for analysis of standard cores, indicates that the TPBAR shapes are bounded by those used for standard plant SBLOCA analysis. The standard core shapes were selected to be limiting with respect to LOCA effects and then scaled to the Technical Specifications limits as required by 10 CFR 50, Appendix K(1)(A).

Swelling and Burst Effects of the TPBARs

As with the LBLOCA, the potential effect of the TPBARs on the adjacent fuel rods during a SBLOCA is through mechanical interaction since they generate negligible heat following the reactor trip and, in particular, during the core uncovery period. In the SBLOCA analysis for the reference plant, the TPBAR temperature in the hot assembly reached 1447°F so that ballooning and burst might be predicted. However, the minimum system pressure during the temperature transient of the rods and TPBAR is 560 psia, well above the levels of a LBLOCA. This pressure together with the TPBAR cladding temperature of less than 1500°F suggests that burst of the TPBARs during a SBLOCA is unlikely. The potential for TPBAR failure during a SBLOCA will be discussed further in Section 3.5.3.

Assuming that rupture did occur in the hot assembly, the mechanical interaction of the TPBARs and the fuel rods would be similar to the situation during the large break. The TPBAR swelling prior to burst is expected to be equal to or less than observed on Zr-4 clad fuel rods. Thus, blockage of the channels adjacent to the thimble should be equal to or less than predicted for a typical fuel rod channel containing 4 fuel rods, and coolable geometry is not a concern.

2.15.5.4 Effect of TPBARs on Post-LOCA Sump Boron Concentration

The post-LOCA containment sump boron concentration is calculated to ensure that sufficient boron exists in the sump to preclude re-criticality when the Safety Injection pumps are switched over from the Refueling Water Storage Tank (RWST) to the sump for cold leg Safety Injection. The calculation is performed for a range of primary system boron concentrations from zero to 1500 ppm and includes all sources of liquid which may reach the containment sump following a LOCA and their respective boron concentrations. It is done to ensure long term core cooling in compliance with the requirements of 10 CFR 50.46(b)(5). Since the TPC core design (Section 2.4.3) does not indicate a need to change the boron concentrations in ECCS components such as the RWST or the Accumulators, insertion of TPBARs does not impact this analysis and the current licensing basis calculations are applicable to the TPBAR core design.

2.15.5.5 Effect of TPBARs on Switchover to Hot Leg Recirculation

The Hot Leg Recirculation Switchover Time analysis is performed to determine the time, following a LOCA, that hot leg recirculation should be initiated in order to preclude boron

precipitation in the reactor vessel. This is done to ensure long term core cooling in compliance with the requirements of 10 CFR 50.46(b)(5) since boron precipitation or boron deposition on the fuel rods would have a detrimental impact on core coolability. Since the TPC core design (Section 2.4.3) does not indicate a need to change the boron concentrations in ECCS components, insertion of TPBARs does not impact this analysis and the current licensing basis calculations are applicable to the TPBAR core design.

2.15.5.6 References

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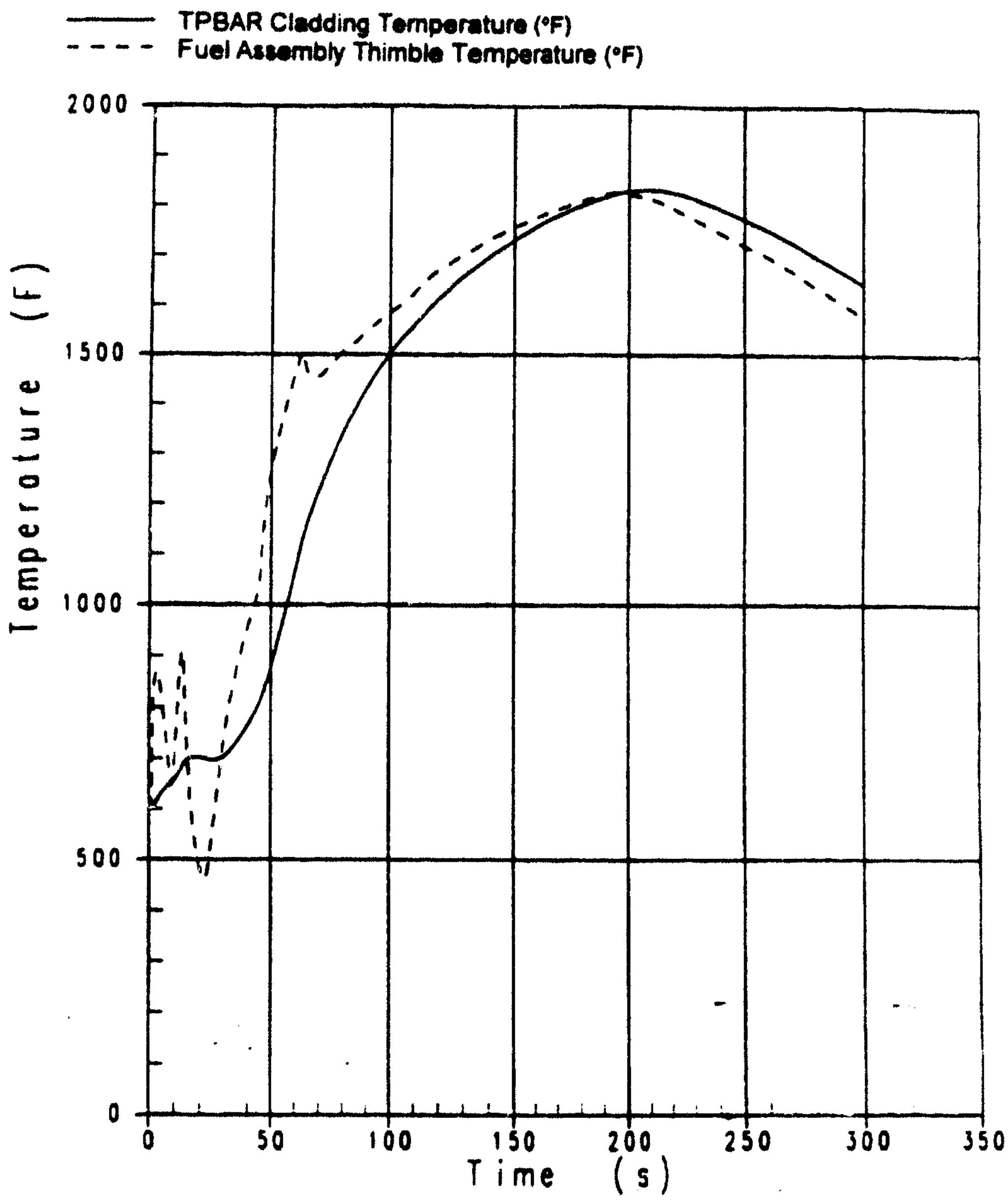

Figure 2.15.5-1
TPBAR Large Break LOCA Peak Cladding Temperature in Reference Plant

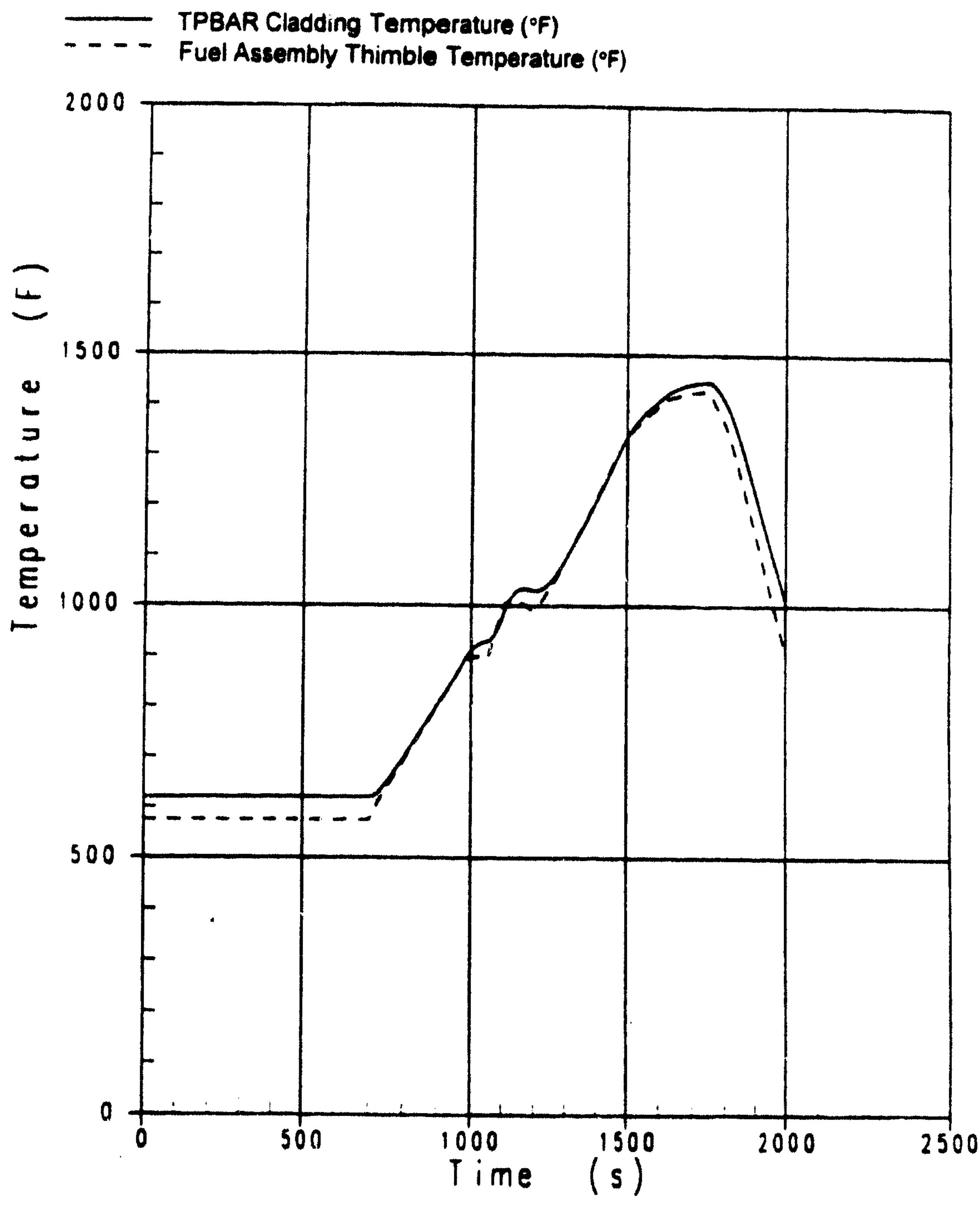

Figure 2.15.5-2
TPBAR Small Break LOCA Peak Cladding Temperature in Reference Plant

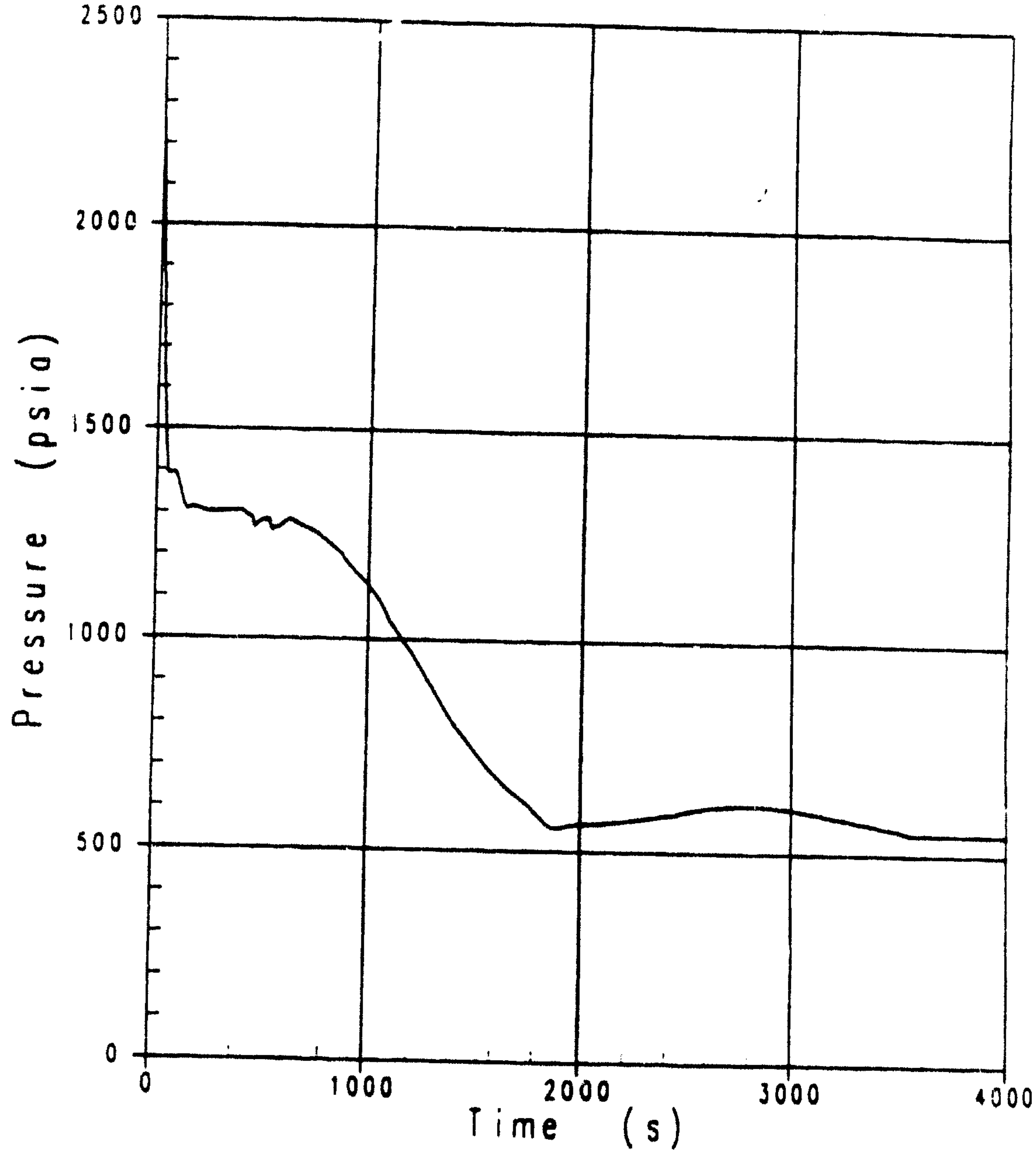

Figure 2.15.5-3
TPBAR Small Break LOCA
Primary System Pressure in Reference Plant

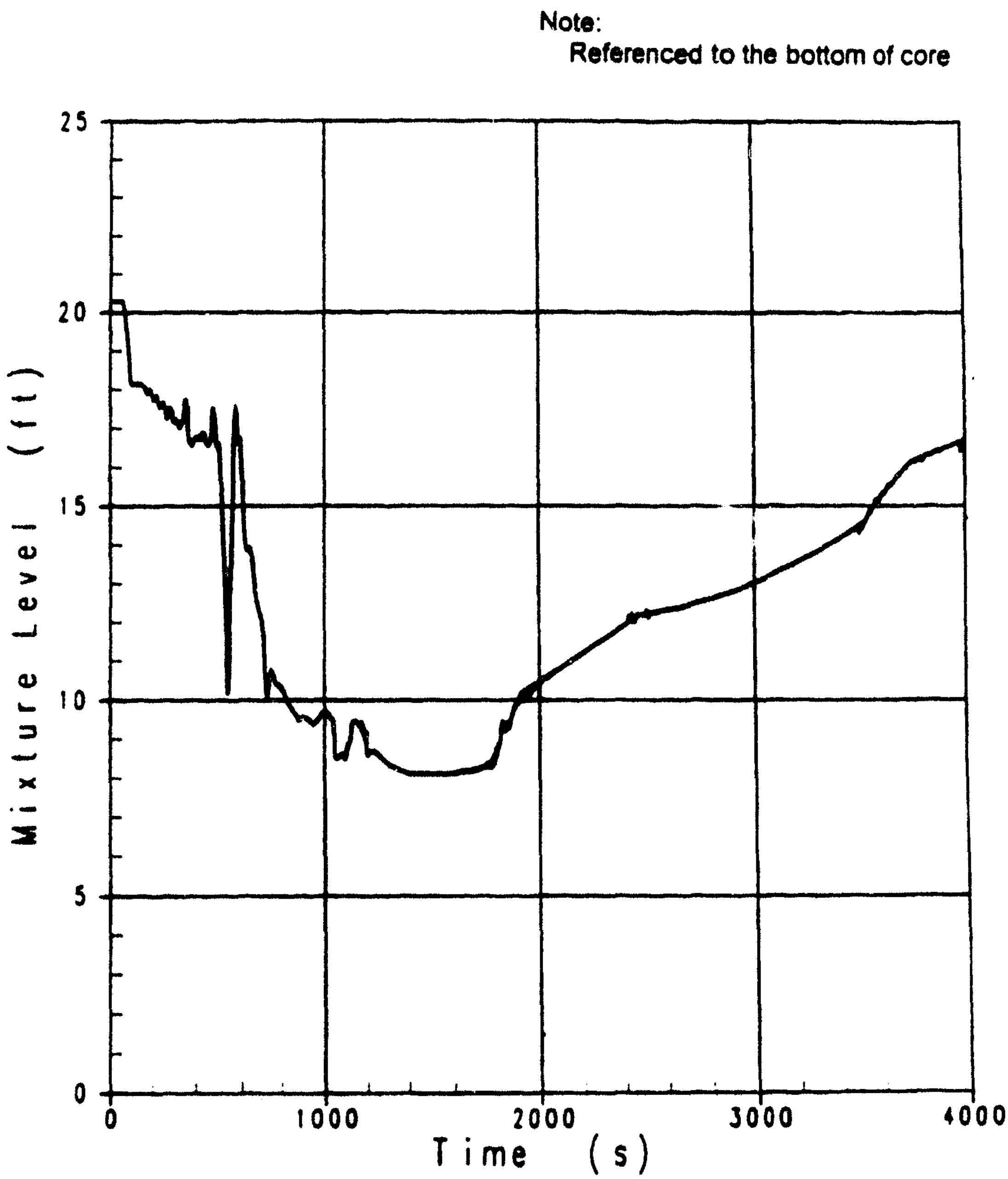

Figure 2.15.5-4
TPBAR Small Break LOCA
Reactor Vessel Mixture Level

2.15.6 RADIOLOGICAL CONSEQUENCES OF ACCIDENTS

2.15.6.1 Introduction

This section addresses the impact of operation with a full core loading of TPBARs on the radiological consequences analyses of the design basis accidents. No TPBAR failures are predicted to occur during the design basis accidents with the exception of the large-break LOCA and the fuel handling accident.

The radiological consequences of accidents are impacted by operation with TPBARs in the core primarily by the addition of the tritium from the TPBARs to the accident source term and also from the fact that the core source term is somewhat different than for operation without TPBARs.

For the reference plant the radiological consequences are determined for the following accidents:

Loss-of-Coolant Accident (LOCA)

Fuel Handling Accident

Reactor Coolant Pump Rotor Seizure

Rod Ejection Accident

Main Steam Line Break

Steam Generator Tube Rupture

Break of Small Line Carrying Primary Coolant Outside Containment

Waste Gas System Failure

Liquid Waste System Failure

The offsite radiological consequences for the reference plant design basis accidents are evaluated against dose limits identified by the NRC in the SRP for thyroid and whole-body (acute dose). For the control room dose evaluation, the SRP specifies a beta-skin dose limit in addition to the limits for thyroid and whole-body.

2.15.6.2 Loss-of-Coolant Accident Consequences (SRP 15.6.5)

The radiological consequences of a LOCA are determined based on the prescriptive assumption that the core cooling is not maintained and that core melting occurs so as to release a large fraction of the core fission product activity. In addition to the core activity releases of 100 percent of the noble gases and 50 percent of the iodines, it is assumed that 100 percent of the tritium in the TPBARs is released to the containment. Table 2.15.6-1 compares the end-of-cycle core source term for iodines and noble gases for operation with and without TPBARs, and it is seen that there is a small decrease in iodine inventories and a small increase in noble gas inventories. A bounding value of 0.9 grams of tritium (8700 Ci) is assumed per TPBAR at the end of the fuel cycle for a total core inventory of 2.91E7 curies.

In modeling the release of tritium to the environment, it is conservatively assumed that the tritium exists solely in the form of tritiated water. This reflects the fact that elemental tritium would relatively quickly exchange with the hydrogen in water to make this a reality (especially considering that the containment is filled with steam and there is ongoing containment spray during the first two hours or longer). Additionally, elemental tritium does not have a significant dose impact since it is not readily absorbed in the body on inhalation while in the form of tritiated water it is easily absorbed into the body and can have a significant impact on dose. With this assumption, most of the tritium will be in the sump solution and only about three percent of the tritium is available for leakage from the containment. When considering Emergency Core Cooling System (ECCS) leakage, this maximizes the tritium that would be available for release from that pathway.

Both the containment leakage pathway and the ECCS leakage pathway contribute to activity releases. The containment leakage pathway releases iodines, noble gases and tritium to the environment and the ECCS leakage pathway releases recirculating sump solution to the auxiliary building. There are no noble gases in this water. All of the iodine in the flashed portion of the water is assumed to become airborne plus ten percent of the iodine in the non-flashed portion. The iodine release from ECCS leakage is reduced by filters with 90% removal efficiency. All of the tritium in the recirculation leakage is assumed to become airborne due to eventual evaporation of the water, and the release of tritium is not affected by filters.

The offsite doses are only slightly changed from those calculated for operation without TPBARs. As shown in Table 2.15.6-2, the whole-body doses are somewhat increased and the thyroid doses are reduced by a small amount. The increase in whole-body doses is due entirely to the change in the core noble gas source term. Tritium does not have any gamma radiation emission and thus does not contribute to the acute dose. Tritium releases contribute to thyroid dose, but the increased dose from tritium is less than the decreased dose due to iodine.

Activity entering the control room is subject to filtration which removes 99% of iodine. Additionally, the recirculation flow has a 99% filtration efficiency which further reduces the control room iodine activity. The noble gases and tritium are not affected by filtration. The doses to the operators in the control room are provided in Table 2.15.6-2. The control room thyroid dose for the reference plant, when operating without TPBARs in the core, has significant margin to the NRC's Standard Review Plan acceptance limit of 30 rem thyroid. The whole-body and beta-skin doses in the control room have only small margins to the NRC's Standard Review Plan acceptance limits of 5 rem whole-body and 75 rem beta-skin. With TPBARs in the core, there is a large increase in the thyroid dose and a small increase in the whole-body dose such that the dose acceptance limits are exceeded. There is a small decrease in the beta-skin dose. The release of tritium to the environment results in a large additional dose to the thyroid, most from the ECCS leakage pathway and the fact that the reference plant has a specified ECCS recirculation leak rate of 2 gpm. If a leak rate typical of most other plants was being used (e.g., 1 gallon per hour), the total thyroid dose from tritium released by ECCS leakage would be reduced by a factor of approximately one hundred and the control room thyroid dose limit would not be challenged. The large control room whole-body and beta-skin doses calculated for the reference plant are due to several factors. One is that the control room is quite large which increases the activity cloud dimensions and this results in a higher gamma field from the airborne activity. Another factor is that there is a high rate of air inflow during

emergency operation which increases the control room doses. Lastly, the reference plant atmospheric dispersion factors for the control room intake are fairly high - a plant with lower dispersion factors would have lower doses. A plant-specific analysis for doses is required, since dose results are plant-specific.

2.15.6.3 Fuel Handling Accident Consequences (SRP 15.7.4)

For the reference plant it is assumed that in a fuel handling accident all the rods in a dropped assembly plus 20% of the rods in an impacted assembly are damaged such that the activity in the fuel/clad gap is released into the pool of water. Thus, it is appropriate to assume that the 24 TPBARs in the dropped fuel assembly plus 5 more in the impacted assembly would also be damaged and would release any free tritium. The maximum tritium buildup is limited to less than 1.2 grams (11,600 curies) in each TPBAR. However, most of the tritium would be retained in the getter portion of the TPBAR and only the tritium in the pores of the pellets (tens of curies per TPBAR) would be free for immediate release into the water pool.

At the spent fuel pool and refueling cavity water temperatures, there would be no significant release of tritium from the getter for an extended period of time (i.e., approximately a year). It is assumed that the damaged TPBARs would be removed and placed into a container before any significant release of tritium from the getter would occur.

Sensitive information has been removed

2.15.6.4 Consequences of Other Design Basis Accidents

Of the remaining design basis events, only the RCP Rotor Seizure (SRP 15.3.3) and the Rod Ejection (SRP 15.4.8) Accidents involve the release of activity from damaged fuel rods. It has been determined that there will be no damage to the TPBARs for these accidents. Thus, the only impact on the radiological consequences is due to the changes in core source terms. These have been evaluated as resulting in a decrease in thyroid dose of approximately 5% and an increase in whole body dose of approximately 10%. There is sufficient margin to dose acceptance limits for the whole body doses that the identified increase in dose is not significant.

For accidents without fuel damage such as Steam Generator Tube Rupture (SRP 15.6.3), Main Steam Line Break (SRP 15.1.5), Small Line Break Outside Containment (SRP 15.6.2), and postulated failures of the gaseous or liquid waste processing systems, there is also no TPBAR damage. For these accidents there is no impact on thyroid doses as a result of operation with TPBARs in the core, since coolant iodine concentrations are limited by the Technical Specifications. The reactor coolant noble gas source terms are affected by operation with TPBARs as indicated in Table 2.15.6-3. The changes in coolant activity result in an increase in whole-body doses of about six percent. The whole-body doses remain within the dose acceptance limits.

Operation with TPBARs does result in an increase in the tritium released to the primary coolant but this increase is countered by an increased discharge of primary coolant to prevent the tritium level in the coolant from exceeding the current operating level. Since coolant tritium levels are not expected to increase, there is no adverse impact on the postulated liquid tank failure releasing water-borne activity to the ground water (SRP 15.7.3).

2.15.6.5 Impact of TPBAR Failure

In the event that a TPBAR suffers cladding degradation late in the operating cycle, it is assumed that the full inventory of tritium could be released to the reactor coolant. Based on a maximum inventory of 11,600 curies of tritium in the rod, this would increase the reactor coolant concentration by 50 μCi/g to 54 μCi/g. If all of this tritium were to be released to the environment, it would add less than 0.05 rem to the offsite thyroid doses.

2.15.6.6 Conclusions

Operation with TPBARs in the core has no adverse impact on offsite thyroid doses but offsite whole body doses increase by 6 to 10 percent over the doses calculated for operation without the TPBARs in the core. The increases in whole body doses are not the result of releases from the TPBARs but are due to changes in the noble gas source term in the core and in the primary coolant associated with revised fuel management. The increases in whole body doses do not challenge dose acceptance limits for any of the design basis accidents.

With TPBARs in the core, the control room beta-skin dose calculated for the Loss-of-Coolant Accident decreases slightly but there are increases in the whole-body dose and the thyroid dose resulting in the dose acceptance criteria being exceeded. In order to achieve an acceptable control room thyroid dose for the reference plant with TPBARs in the core, the permissible post-LOCA recirculation leakage would have to be significantly reduced from the current value of 2.0 gpm (other plants have recirculation leakage specified at near one gallon per hour). An acceptable control room whole-body dose could be obtained for the reference plant with TPBARs in the core if the containment leak rate were reduced slightly from the current value of 0.2% per day. Other plants have a more substantial margin to the dose acceptance limit and would not require any change in the containment leak rate to attain an acceptable whole-body dose in the control room.

Table 2.15.6-1
Core Iodine and Noble Gas Source Terms For Design Basis Radiological Consequences Analysis

	Core without TPBARs (Ci)	Core with TPBARs (Ci)
I-131	9.87E7	9.31E7
I-132	1.44E8	1.35E8
I-133	2.02E8	1.94E8
I-134	2.17E8	2.14E8
I-135	1.87E8	1.81E8
Kr-85m	2.58E7	2.77E7
Kr-85	8.89E5	6.86E5
Kr-87	4.74E7	5.40E7
Kr-88	6.75E7	7.62E7
Xe-131m	6.66E5	9.45E5
Xe-133m	2.89E7	5.99E6
Xe-133	1.98E8	1.94E8
Xe-135m	4.02E7	3.71E7
Xe-135	4.47E7	4.70E7
Xe-138	1.62E8	1.63E8

Table 2.15.6-2 Radiological Consequences Of A Design Basis LOCA (rem)			
	Operation without TPBARs	**Operation with TPBARs**	**Acceptance Limit**
Site Boundary			
Thyroid dose			
- Containment leakage	62.4	59.2	
- Recirculation leakage	2.34	2.23	
Total	64.7	61.4	300
Whole body dose			
- Containment leakage	1.6	1.7	
- Recirculation leakage	0.0063	0.0061	
Total	1.6	1.7	25
Low Population Boundary			
Thyroid dose			
- Containment leakage	69.0	65.4	
- Recirculation leakage	9.38	9.05	
Total	78.4	74.4	300
Whole body dose			
- Containment leakage	1.1	1.2	
- Recirculation leakage	0.0094	0.009	
Total	1.1	1.2	25
Control Room			
Thyroid dose			
- Containment leakage	18.9	18.4	
- Recirculation leakage	1.94	44.1[1]	
Total	20.9	62.5	30
Whole body dose			
- Containment leakage	4.8	5.1	
- Recirculation leakage	6.2E-5	6.0E-5	
Total	4.8	5.1	5
Beta-skin			
- Containment leakage	67.2	62.4	
- Recirculation leakage	5.2E-4	2.5	
Total	67.2	64.9	75[2]

Notes:

1. The large increase in thyroid dose is due to the release of tritium associated with the high rate of leakage of post-LOCA sump solution defined for the reference plant. The fraction due to tritium is 0.96.

2. This limit is that associated with taking credit for protective clothing.

Table 2.15.6-3 Primary Coolant Noble Gas Source Term for Operation with 1.0% Fuel Defects		
	Operation without TPBARs (μCi/g)	Operation with TPBARs (μCi/g)
Kr-85m	1.964	1.905
Kr-85	7.136	7.352
Kr-87	1.235	1.232
Kr-88	3.540	3.603
Kr-89	0.104	0.0973
Xe-131m	1.883	3.017
X3-133m	16.94	4.677
Xe-133	238.9	267.7
Xe-135m	0.542	0.450
Xe-135	7.979	8.257
Xe-138	0.711	0.607

2.15.7 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

2.15.7.1 Introduction

An anticipated transient without scram (ATWS) is an anticipated operational occurrence during which an automatic reactor scram is required, but fails to occur due to some common mode fault in the reactor protection system. Under certain circumstances, failure to execute a required scram during an anticipated operational occurrence could transform a relatively minor transient into a more severe accident. ATWS events are not considered to be in the design basis for plants designed by Westinghouse, including the reference plant with the TPC.

For Westinghouse plants, a series of generic studies (References 2 and 3) on ATWS has shown that acceptable consequences will result, provided that the turbine is tripped and auxiliary feedwater flow is initiated within Technical Specification limits. The limiting criterion associated with the ATWS analyses for Westinghouse plants is that the maximum RCS pressure does not exceed 3200 psig, which is the pressure corresponding to the ASME Service Level C stress limit in the most stress limiting RCS component. Consistent with the results of the generic studies, the final NRC ATWS rule (Reference 4) requires that Westinghouse-designed plants install ATWS mitigation system actuation circuitry (AMSAC) to initiate a turbine trip and actuate auxiliary feedwater flow, independent of the reactor protection system. The reference plant AMSAC design is described in Section 7.7 of the reference plant FSAR (Reference 1).

One of the plant conditions important to demonstrating acceptable consequences for any specific ATWS event is the net amount of negative reactivity feedback available. In lieu of using the control rods to shut down the reactor, the core must rely upon inherent feedback mechanisms (Doppler and moderator) to aid in bringing the reactor under control. Consideration of the total reactivity feedback produced by the interaction of the Doppler and moderator temperature coefficients during the ATWS transient is reflected in the calculation of unfavorable exposure time (UET) using the critical power trajectory methodology that has been presented by Westinghouse in Reference 5, for application to the ATWS event.

2.15.7.2 Method of Analysis

The critical power trajectories are the calculated loci of plant conditions (e.g., power versus inlet temperature) which produce a peak pressure for the limiting ATWS event that matches the specified peak RCS pressure limit of 3200 psig. The UET is the time during the cycle when the total reactivity feedback is insufficient to maintain the RCS pressure under the 3200 psig limit for a given reactor state. In the process of calculating the UET for a core, the ATWS transient point kinetics information is converted into steady state conditions and the critical power trajectories are determined. Cycle specific steady state core design calculations are performed with the appropriate ATWS initial conditions of full power, rods out, equilibrium xenon, and 3200 psig RCS pressure. Criticality is determined as a function of core inlet temperature and these results are compared directly to the critical power trajectory from the transient analyses. This comparison provides cycle specific reactivity conditions that would result in the subject transient exceeding the 3200 psig limit. Further calculations as a function of core burnup explicitly quantify the fraction of time during the cycle that the core design fails to meet

reactivity feedback limits defined by the transient based critical power trajectories. From this, the UET is determined.

2.15.7.3 Results

Analysis has been performed to define the impact of the TPBARs on the performance of the reference plant with respect to the limiting ATWS event. Critical power trajectory curves that apply to the reference plant have been used in conjunction with results from explicit steady state nuclear design calculations for the Cycle 1 and equilibrium cycle core designs described in Section 2.4.3. The predicted UETs for both analyzed TPC cycle designs are less than that for the reference plant core design. Therefore, the TPC cycle designs described in Section 2.4.3 of this report actually improve the response of the reference plant to the limiting ATWS event.

The reason that the predicted UETs for the TPC core designs are somewhat better than for the reference plant is primarily due to the relative magnitude of the respective moderator temperature coefficients (MTCs). In the TPC designs, a large number of burnable absorbers (both IFBA and TPBAR) are used to control the excess core reactivity at BOL to maintain reasonable critical boron concentrations. The excess core reactivity for the TPC designs is larger than for typical reference plant cycle designs because more excess core reactivity is needed to achieve the required cycle energy in the presence of TPBARs, which are a large reactivity penalty at EOL. The large burnable absorber inventory tends to reduce the boron worth at BOL (and the boron concentration in the TPC first cycle), which causes the MTC to be more negative early in the cycle relative to typical reference plant reload cores. This, in turn, reduces the critical powers and the UET.

Reload cores which employ large numbers of TPBARs will always tend to have more negative MTCs than standard reference plant reload cores early in life, as long as the cycle energy is comparable to typical reference plant cores and the critical boron concentration is controlled to typical values with IFBA. This type of fuel management leads to lower boron worths, more negative MTCs, and therefore, more favorable UETs for the TPBAR cores. Section 2.4.3 of this report (Nuclear Design) includes a figure that plots the hot full power MTC as a function of core burnup, for the reference plant core, the Cycle 1 TPC, and the equilibrium TPC.

2.15.7.4 Conclusions

Both the Cycle 1 and equilibrium cycle TPC designs reduce the predicted UET for the limiting ATWS event, compared to the reference plant core design. Therefore, the TPC designs described in this report are less limiting than the reference plant core design with respect to the ATWS event.

2.15.7.5 References

1. Reference Plant, Final Safety Analysis Report Update

2. WCAP-8330, "Westinghouse Anticipated Transients Without Trip Analysis," August 1974.

3. Anderson, T. M., "ATWS Submittal," Westinghouse Letter NS-TMA-2182 to S. H. Hanauer of the NRC, December 1979.

4. ATWS Final Rule - Code of Federal Regulations 10 CFR 50.62 and Supplementary Information Package, "Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

5. WCAP 11992, "Joint Westinghouse Owners Group/Westinghouse Program: ATWS Rule Administration Process," December 1988.

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2.16 TECHNICAL SPECIFICATIONS

2.16.1 INTRODUCTION

There is only one review plan in this chapter of the SRP. The acceptance criteria in SRP 16.0 are based on the relevant requirements of 10 CFR 50.34 and 50.36. They deal with the preparation of technical specifications for the plant consistent with the regulatory guidance contained in the Standard Technical Specifications developed for plants by a particular NSSS vendor. The latest Standard Technical Specifications for Westinghouse-supplied NSSS plants are contained in NUREG-1431, Rev. 1, "Standard Technical Specifications Westinghouse Plants." NUREG-1431, and the reference plant Technical Specification (which are based on NUREG-1431) have been reviewed for potential changes due to incorporation of TPBARs. The potential changes which were identified are discussed below, along with potential changes to the Bases sections. No other Technical Specification changes are anticipated.

2.16.2 POTENTIAL TECHNICAL SPECIFICATION CHANGES

One Technical Specification which may need to be revised is the Spent Fuel Assembly Storage (TS 3.7.17 in NUREG-1431) and its accompanying "Design Features" description (TS 4.3, Fuel Storage). TS 3.7.17 contains a figure which defines acceptable combinations of fuel assembly burnup and initial U-235 enrichment for a particular fuel storage configuration. TS 4.3 for the reference plant describes in detail the allowed spent fuel storage configurations. As discussed in Section 2.9.2.2 of this report, the spent fuel storage criticality will need to be addressed on a plant-specific basis. Therefore, when an applicant prepares a License Amendment Request to incorporate TPBARs, the Fuel Storage Technical Specifications should be reviewed and revised if necessary.

As discussed in Sections 2.5.2 and 2.5.3 of this report, the Cold Overpressure Mitigation System setpoints, and the heatup and cooldown pressure-temperature limit curves will probably be impacted by the incorporation of TPBARs. For some older plant Technical Specifications, these would be specified in TS 3.4.12 and 3.4.3. However, both NUREG-1431 and the reference plant Technical Specifications reference the PTLR for these values. Therefore, plants which have a PTLR will need to revise it, but will not need to revise TS 3.4.12 or 3.4.3.

2.16.3 POTENTIAL TECHNICAL SPECIFICATION BASES CHANGES

The Fuel Storage Technical Specification Bases (B 3.7.17 in NUREG-1431) may need to be revised, depending on the plant-specific evaluation of criticality, as discussed in Section 2.16.2, above.

The Bases for the Hydrogen Recombiners Technical Specification (B 3.6.7) contains a discussion in the "Applicable Safety Analyses" section which lists the sources of hydrogen in containment following a LOCA. This Bases should be revised to include the two additional sources of hydrogen which were discussed in Section 2.6.5 of this report.

Based on the discussion in Section 2.9.4, a plant-specific evaluation will need to be performed to determine the impact of the increased spent fuel pool heat load on the Component Cooling Water (CCW) System and its ability to provide cooldown within a specified time. The Bases section for the CCW Technical Specification (B 3.7.7) for the reference plant explicitly defines a time limit for meeting the cold shutdown requirements, although NUREG-1431 does not. Applicant plants who wish to incorporate TPBARs will need to assess the impact of the increased spent fuel pool heat load on the cooldown, and review the CCW Technical Specification Bases to determine whether a revision is necessary.

2.17 QUALITY ASSURANCE

2.17.1 INTRODUCTION

Chapter 17 of the SRP deals with the Quality Assurance controls applicable during all phases of a facility's life. Section 2.17.2, below, describes the Quality Assurance programs which are applicable to aspects of the TPBAR incorporation into the reference plant.

SRP 17.1
Quality Assurance During the Design and Construction Phases: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix B; 10 CFR 50, Appendix A; 10 CFR 50.55a; 10 CFR 50.55(e); and 10 CFR 50.34(a.7) with emphasis on activities associated with the design and construction phases. They deal with the QA controls related to the 18 areas outlined in 10 CFR 50, Appendix B. Additional regulatory guidance is provided in the following documents: Reg Guides 1.8, 1.26, 1.28, 1.29, 1.30, 1.37, 1.38, 1.39, 1.58, 1.64, 1.74, 1.88, 1.94, 1.116, 1.123, 1.144, 1.146, and Branch Technical Position CMEB 9.5-1.

SRP 17.2
Quality Assurance During the Operations Phase: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix B; 10 CFR 50, Appendix A; 10 CFR 50.55a; and 10 CFR 50.34(b.6ii) with emphasis on activities associated with the operations phase (including maintenance and modification). They deal with the QA controls related to the 18 areas outlined in 10 CFR 50, Appendix B. Additional regulatory guidance is provided in the following documents: Reg Guides 1.8, 1.26, 1.29, 1.30, 1.33, 1.37, 1.38, 1.39, 1.58, 1.64, 1.74, 1.88, 1.94, 1.116, 1.123, 1.144, 1.146, and Branch Technical Position CMEB 9.5-1.

SRP 17.3
Quality Assurance Program Description: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix B; 10 CFR 50, Appendix A, GDC 1; 10 CFR 50.55a; 10 CFR 50.55(e); and 10 CFR 21. The QA program is organized into three discrete areas of activity: 1) management, 2) performance/verification, and 3) self-assessment. These three areas encompass the eighteen areas defined in 10 CFR 50, Appendix B. The QA program applies to all phases of a facility's life, including design, construction, operation, modification, and decommissioning. Additional regulatory guidance is provided in the following documents: Reg Guides 1.8, 1.26, 1.28, 1.29, 1.33, 1.152, 1.143, 1.36, 1.54, and 4.15, Generic Letter 89-02, and Branch Technical Position CMEB 9.5-1.

2.17.2 QUALITY ASSURANCE PROGRAM

Westinghouse Energy Systems is a vendor providing items and services for commercial nuclear power plants in compliance with the requirements of 10 CFR 50, Appendix B as prescribed in Regulatory Guide 1.28. The Westinghouse Quality Management System (QMS) describes the policies and commitments of Energy Systems quality assurance program. The NRC has reviewed the QMS, R. 2, and concluded that it meets the requirements of 10 CFR 50, Appendix

B, and could be utilized to control the quality of items and services provided by Energy Systems. The QMS complies with SRP 17.1.

Design and manufacturing activities for commercial nuclear plants are controlled in accordance with the QMS as implemented by Energy Systems procedures, division-specific procedures and additional lower level procedures. Design activities for the TPC have been and will be controlled similarly, with TPC-specific implementing procedures created when necessary. TPBAR design activities are undertaken at PNNL in accordance with their QA system which has been reviewed and approved by Westinghouse. PNNL is approved by Westinghouse under the Westinghouse QMS as a qualified supplier of TPBAR design information. When manufacturing activities are undertaken, the same quality assurance approach will be applied.

SRP 17.2 is a guide for the review and acceptance of quality assurance controls applied during the operation, maintenance and modification of a nuclear power plant. These controls are plant-specific and the acceptability of such quality assurance controls under the SRP for the TPC needs to be addressed on a plant-specific basis.

SRP 17.3 is a guide for the review and acceptance of new quality assurance programs received after SRP Section 17.3 was noticed in the Federal Register. For previously existing NRC-accepted quality assurance programs reviewed using SRP 17.1, SRP 17.1 will continue to be used for quality assurance program description review unless the submitter specifically requests SRP 17.3 to be applied. The Energy Systems quality assurance program existed prior to the Federal Register notice of SRP 17.3, and Energy Systems has not requested SRP 17.3 to be applied; therefore, SRP 17.3 is not applicable.

2.18 HUMAN FACTORS ENGINEERING

The sections of Chapter 18 of the SRP deal with the adequacy of the control room design and the Safety Parameter Display System, relative to human factor considerations. Based on the following qualitative discussions, it was determined that evaluations of potential TPBAR impact were not required, and it is not anticipated that there would be changes to Chapter 18 of a typical SAR as a result of incorporating TPBARs.

SRP 18.1
Control Room: The acceptance criteria in this section are based on the relevant requirements of Task Action Plan Item I.D.1 of NUREG-0660 as clarified in Supplement 1 of NUREG-0737, for licensees with existing control rooms. They deal with the identification of tasks to be accomplished by operators in the main control room and remote shutdown areas under emergency operating conditions; the provision in the control room and remote shutdown areas of information, displays, controls and other interfaces necessary for operators to successfully carry out these tasks; and the design of these control stations such that personnel responsible for operating the plant from the control room and remote shutdown areas can perform their tasks in as error-free and timely a manner as possible. The incorporation of the TPBARs does not require additional emergency procedures to be carried out, and the control room and remote shutdown area designs are not being changed, therefore, there is no impact.

SRP 18.2
Safety Parameter Display System (SPDS): The acceptance criteria in this section are based on the relevant requirements of Task Action Plan Item I.D.2 of NUREG-0660 as clarified in Supplement 1 of NUREG-0737, for licensees with existing control rooms. They deal with the location of the SPDS such that it is convenient to control room operators; the continuous display of information from which control room personnel can readily and reliably assess the safety status of the plant; the concise display of plant variables which provide information to plant operators about reactivity control, reactor core cooling and heat removal from the primary system, reactor coolant system integrity, radioactivity control, and containment conditions; and the incorporation of human factors principles so that displayed information can be readily perceived and comprehended. The incorporation of the TPBARs does not require any additional displays on SPDS, and the design of the current SPDS is not being changed, therefore, there is no impact.

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SECTION 3 TPC TPBAR EVALUATION

3.1 INTRODUCTION

This section of the Topical Report includes a description of the design, the design basis, and the performance evaluation of tritium producing burnable absorber rods (TPBARs) to be used in reload fuel assemblies of large commercial PWRs. The functions of TPC TPBARs, other than producing and retaining tritium, are comparable to those of burnable absorber rods used in commercial PWRs for fuel cycle reactivity control. The TPC TPBARs have been designed to be compatible with fuel designs in a 17 x 17 fuel array. The conclusion from this and previous evaluations is that the TPBAR nuclear, mechanical and thermal characteristics are comparable and compatible with those of conventional burnable absorber rods (Reference 1 and 2). The TPBARs have only one safety function: to perform their neutron absorption function in conjunction with the reactivity control system.

In this section of the report, the performance of the getter-barrier type TPBARs is evaluated for a tritium production core loaded with ~ 3344 TPBARs. The design conditions envelope large PWR core conditions with core power density equal to or less than the reference core design. The tritium production, mechanical and thermal performance design conditions are selected to be generic for TPBAR operation in a large PWR core with the range of characteristics described in this report.

The TPBAR design and performance was evaluated in Technical Report PNNL-TTQP-6-503, Rev. 1 (Reference 1), for a lead test assembly (LTA) in the TVA Watts Bar Nuclear Power Plant (WBNP). This technical report was reviewed by the NRC with conclusions documented in a Safety Evaluation Report, Project Number 697, in April 1997 (Reference 3). Remaining issues from the review of Reference 3 were resolved and Reference 4 authorized TVA to irradiate the LTAs. The evaluation of the TPC TPBAR design for this tritium production topical report builds on these documents. Specifically, many details of the LTA TPBAR evaluation are valid and applicable to a generic TPBAR design and will be invoked by reference to the previous reports. The differences between the LTA TPBAR and the TPBAR designed for the TPC (also called TPC TPBAR) are:

- a partial length absorber (^{6}Li) pellet stack not extending over the full core length,
- a greater ^{6}Li loading, and
- a modified weld joint design.

The TPBARs irradiated in the LTA in WBNP were designed to operate in a non-lead position of the Cycle 2 core of that reactor. With this limitation, the operating condition for the LTA in WBNP and the TPBARs in the TPC are different. However, the design requirements and design conditions assumed for the LTA are essentially the same as those for the TPC TPBARs. To highlight this, the LTA design and operating parameters are listed with the TPC TPBAR values for comparison, where appropriate. In addition, design feature differences of TPC and LTA TPBARs are addressed. The TPBAR with the modifications for the TPC is also named TPC TPBAR to distinguish it from the LTA TPBAR.

The mechanical and thermal analyses for enveloping and reference design conditions of the TPC TPBAR are performed for the range of design conditions specified by the reference TPC core design. The TPC TPBAR is compatible with the fuel assembly, reactor vessel internals, reactor coolant chemistry, and refueling system and tools of the reference plant. The TPC

TPBAR assembly is a removable reactor core component that resides in a fuel assembly. TPBARs will not reside in fuel assemblies that contain a rod cluster control assembly.

The TPC TPBAR design ensures that: a) TPBARs will not be damaged nor will they damage the fuel system during Condition I or Condition II events; b) TPBARs will not adversely impact control rod insertion; c) TPBARs will not adversely impact reactor core coolability; and d) tritium release shall be within acceptable limits. During Conditions III and IV, except for the LBLOCA, the TPBARs maintain their structural integrity.

This report is submitted by the DOE to the NRC with the expectation that the NRC will review the report, document the findings in a Safety Evaluation Report (SER) and permit the use of TPBARs in a TPC. Suppliers of tritium production core loads and utilities can reference the topical report in license amendment requests for a tritium production core in a PWR to the extent specified and within the limitations delineated in this report. The SER provides the basis for acceptance of the report and implementation of TPBARs in large PWRs for tritium production. This report will be referenced in license amendment requests in conjunction with an evaluation that the conclusion of this topical report is applicable to the specific plant and fuel cycle involved. Only designs, materials and conditions addressed in the topical report can be referenced as applicable in the license amendment requests.

3.1.1 REFERENCES

1. PNNL-TTQP-6-503, Rev 1 (CRD), "Report on the Evaluation of the Tritium Producing Burnable Absorber Rod Lead Test Assembly", February 1997.

2. NP-1974, S. M. Stoller Corporation, "Control Rod Materials and Burnable Poisons", November 1981.

3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Department of Energy's proposal for the irradiation of the lead test assemblies containing tritium producing burnable absorber rods in commercial light water reactors", Project No. 697, April 1997.

4. Letter R. E. Martin, NRC to O. D. Kingsley, Jr., TVA, "Issuance of Amendment for Tritium Producing Burnable Absorber Rod Lead Test Assemblies" (TAC NO.M98615), September 15, 1997.

3.2 TPC TPBAR DESIGN

3.2.1 DESIGN DESCRIPTION

The Department of Energy has developed a TPBAR that will be inserted into conventional commercial fuel assemblies (Reference 1). The design and the functions of the TPBAR are similar to those of burnable absorber rods that are used in commercial PWR fuel assemblies to manage the core reactivity (Reference 2). The external dimensions and the austenitic stainless steel cladding of the TPBAR are similar to those of Burnable Poison Rod Assemblies (BPRAs), as shown in Table 3.2-1. However, the TPBARs use lithium-6, instead of boron-10, as neutron absorbing material. Tritium is produced in annular lithium aluminate ($LiAlO_2$) pellets through transmutation in the following (n/α) reaction:

$$^{6}Li\ (n, {}^{4}He)\ {}^{3}H + 4.8\ MeV$$

The amount of ^{6}Li required in the TPBAR to achieve a particular tritium production is dictated by nuclear core design.

The TPBAR design consists of concentric cylindrical components clad with Type 316 stainless steel (316 SS), as illustrated in Figures 3.2-1 and 3.2-2. The 316 SS tubular cladding provides structural strength and is the pressure barrier between the TPBAR internals and the reactor coolant system. The inner surface of the cladding is coated with a permeation-resistant aluminized barrier.

The TPBAR internal components are a plenum subassembly, twelve pencils and a lower getter disc. A pencil consists of a Zircaloy-4 liner around which are stacked absorber pellets that are confined in a getter tube, as illustrated in Figure 3.2-1.

The TPBAR design for the TPC uses thin walled annular $LiAlO_2$ pellets assembled into stacks extending over the full or partial length of the active core. The TPC TPBAR pellet stack length of lithium aluminate pellets enriched in Li-6 is 128.5 inches for the reference equilibrium core and 127.5 inches for a first core, starting approximately 5.4 inches from the bottom end of the TPBAR and approximately 7.5 inches above the bottom of the active fuel pellets in the adjacent fuel rods. Above and below the Li-6 enriched lithium aluminate pellets are short stacks of pellets fabricated from depleted lithium aluminate. Aluminum oxide may be considered as a replacement for the depleted lithium aluminate in future designs. These short pellet stacks are used to position the enriched pellet column within the core. The pellets are packaged into stacks of twelve components, referred to as "pencils". The use of a part length absorber pellet stack is the most significant change from the LTA TPBAR design. Future core designs will use absorber pellet stack length commensurate with power peaking and production requirements. The getter tube surrounds the absorber pellets and is composed of nickel-plated Zircaloy-4 (NPZ).

Loaded into the cladding tube above the pencil stack, are a compression spring and an upper getter assembly. The compression spring is fabricated from stainless steel and is similar in design to the PWR fuel and control rod springs. Its function is to provide an axial force to restrain the pencil stack during handling and loading operations prior to irradiation while allowing thermal axial growth of the stack in the reactor.

Sensitive information has been removed

For closure of the TPBAR, end fittings are welded to each end of the cladding tube. The end fittings are manufactured from 316 SS and attached to Westinghouse BPRA rod end fittings with minor modifications. The upper end plug fitting provides for the attachment of the TPBAR to the holddown assembly in the same manner used to attach the BPRA. Prior to the final closure weld, the TPBAR is evacuated and repressurized to one atmosphere with helium.

The TPBAR dimensions and tolerances are specified to provide sufficient clearances between components for assembly, at beginning of life (BOL) and at end of life (EOL) for normal operations; and to prevent interference with the fuel assembly components. The clearances were established to provide for dimensional changes due to thermal, irradiation, creep and hydriding effects. Properties of the TPBAR component materials used for design and analyses are compatible with the reactor coolant environment and are documented in the Materials Properties Handbook (MPH)(Reference 3).

3.2.2 TPBAR OPERATION

Sensitive information has been removed

The irradiation design base case for the TPC TPBAR is 520 effective full-power days (EFPD). The TPC TPBARs will reside in the reactor core for one fuel cycle and therefore exposure is expected to be less than 494 EFPD.

3.2.3 TPBAR SUPPORT IN THE CORE STRUCTURE

The TPBARs are attached to a core component holddown assembly to allow insertion into and removal from a fuel assembly. The holddown assembly shown in Figure 3.2-3 comprises a base plate, spring guide, holddown spring, a yoke or holddown bar and pins.

The TPBARs are mounted to the base plate of a holddown assembly by a single threaded connection. The threaded top end plug is placed in the mounting hole in the baseplate and held by a nut. [

]^{a,c}

The holddown assembly is designed and was analyzed to support core components such as burnable absorber rods, source rods and thimble plugs. Attaching 24 TPC TPBARs or less does not change the functions of the holddown assembly. The analysis shows that it is designed to support components of a weight greater than or similar to TPBARs, and to provide the capability to handle the assembly with the TPBARs attached. The holddown assembly supports the rods inside the thimble tubes and firmly locates the assembly between the upper core plate and the fuel assembly upper nozzle.

The holddown spring is loaded in compression by the spring yoke that contacts the upper core plate. The spring has the ability to absorb the motion between the core plate and the fuel assembly top nozzle to accommodate differential thermal expansion and irradiation growth. The design prevents the spring from going solid under postulated conditions of the fuel assembly top nozzle pads contacting the upper core plate. The holddown spring provides sufficient force to prevent the base plate from lifting away from the fuel assembly top nozzle adapter plate under the highest anticipated normal hydraulic forces.

The TPBARs are inserted into thimble tubes that are standard for all fuel assemblies in a core. The thimble tubes channel coolant to the TPBARs as shown in Figure 3.2-4. Most of the flow enters the thimble tubes through four flow holes near the bottom of the thimble shown in the figure. A small amount of flow enters the thimble tube through a bottom orifice and cools the TPBAR lower end plug. When the fuel assembly is located under a control rod, the lower end of the thimble tube functions as a dashpot.

3.2.4 REFERENCES

1. PNNL-TTQP-6-503, Rev 1 (CRD), "Report on the Evaluation of the Tritium Producing Burnable Absorber Rod Lead Test Assembly", February 1997.

2. NP-1974, S. M. Stoller Corporation, "Control Rod Materials and Burnable Poisons", November 1981.

3. TTQP-7-008, "Materials Properties Handbook for the Tritium Target Qualification Program", Pacific Northwest National Laboratory, Richland, Washington.

Table 3.2-1 Comparison of TPBAR with Burnable Absorber Rod Design Parameters				
Parameter	**WABA Rod 17x17 FA**	**BPRA Rod 17x17 FA**	**LTA TPBAR**	**TPC TPBAR**
Overall Length, in.	150*	152	152	152
Total Weight, lbs	1.9	1.8	2.26	$\leq$2.23
Absorber Length, in.	134*	142	142	127.5** 128.5 + 144.0++
Absorber O.D., in. Thickness, in.	[ ]^{a,c} 0.020	[ ]^{a,c} 0.073	0.303 0.040	0.303 0.040
Absorber Material	Al_2O_3-B_4C	SiO_2-B_2O_3	$LiAlO_2$	$LiAlO_2$
Outer Cladding O.D., in. Thickness, in.	0.381 [ ]^{a,c}	0.381 [ ]^{a,c}	0.381 0.0225	0.381 0.0225
Inner Cladding O.D., in. Thickness, in.	[ ]^{a,c} [ ]^{a,c}	None	None	None
Cladding Material	Zircaloy-4	304SS	316SS	316SS
Fuel Assembly Guide Thimble I.D., in.	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}
Fuel Assembly Dashpot I.D., in.	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}	[ ]^{a,c}

* Length may vary to accommodate a specific plant application

** First TPC

\+ Equilibrium TPC

++Maximum

Sensitive information has been removed

Figure 3.2-1
TPBAR Longitudinal Cross Section

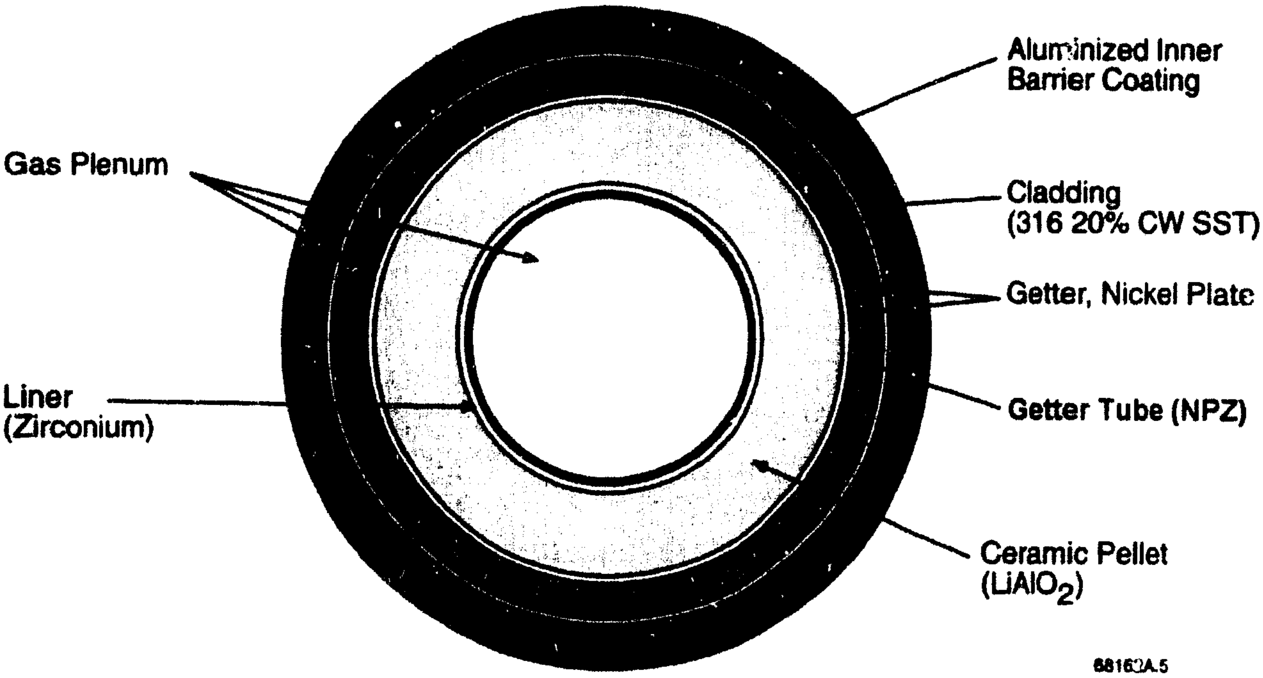

Figure 3.2-2
TPBAR Transverse Cross Section

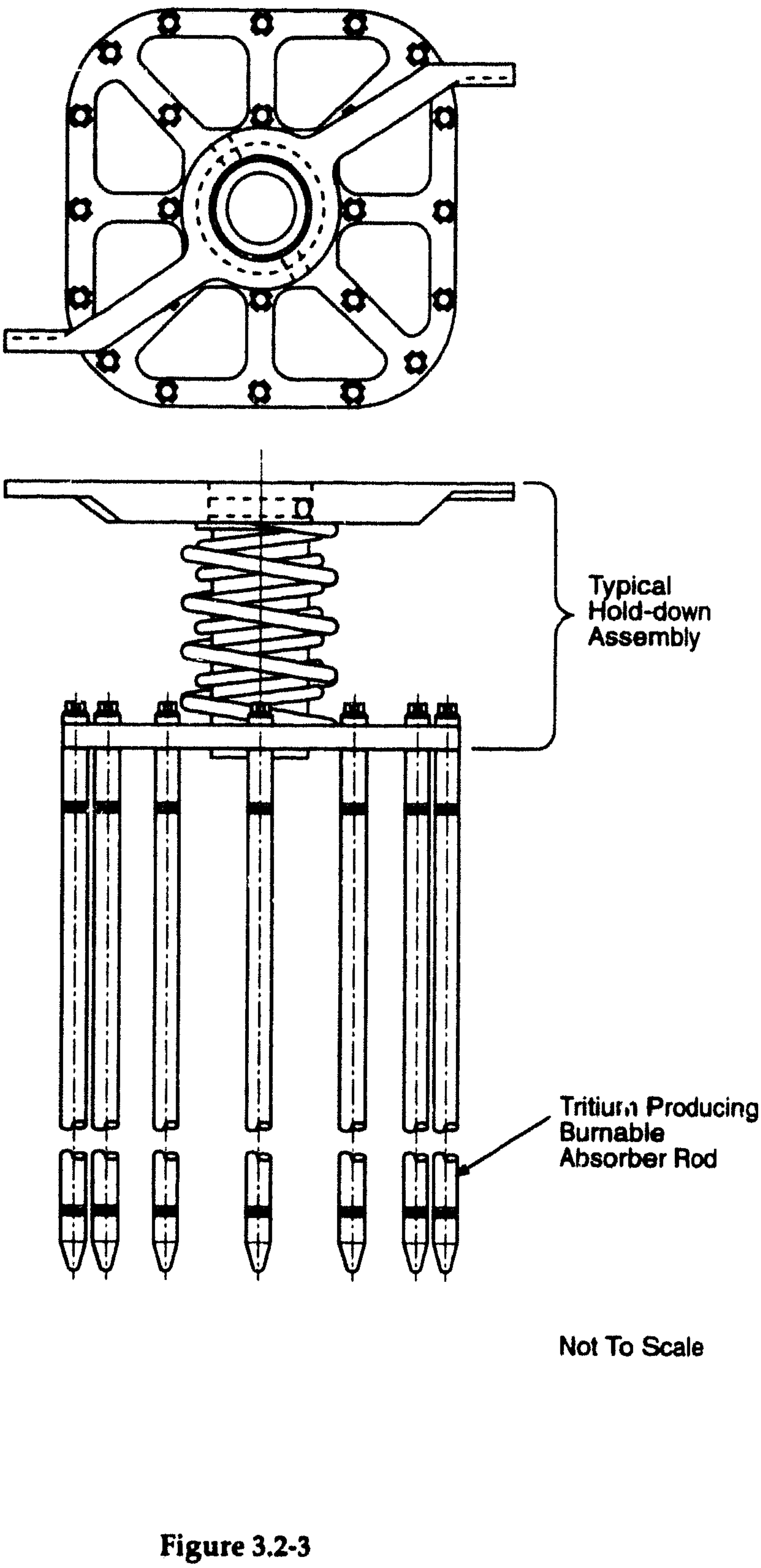

Figure 3.2-3
TPBAR Hold-down Assembly

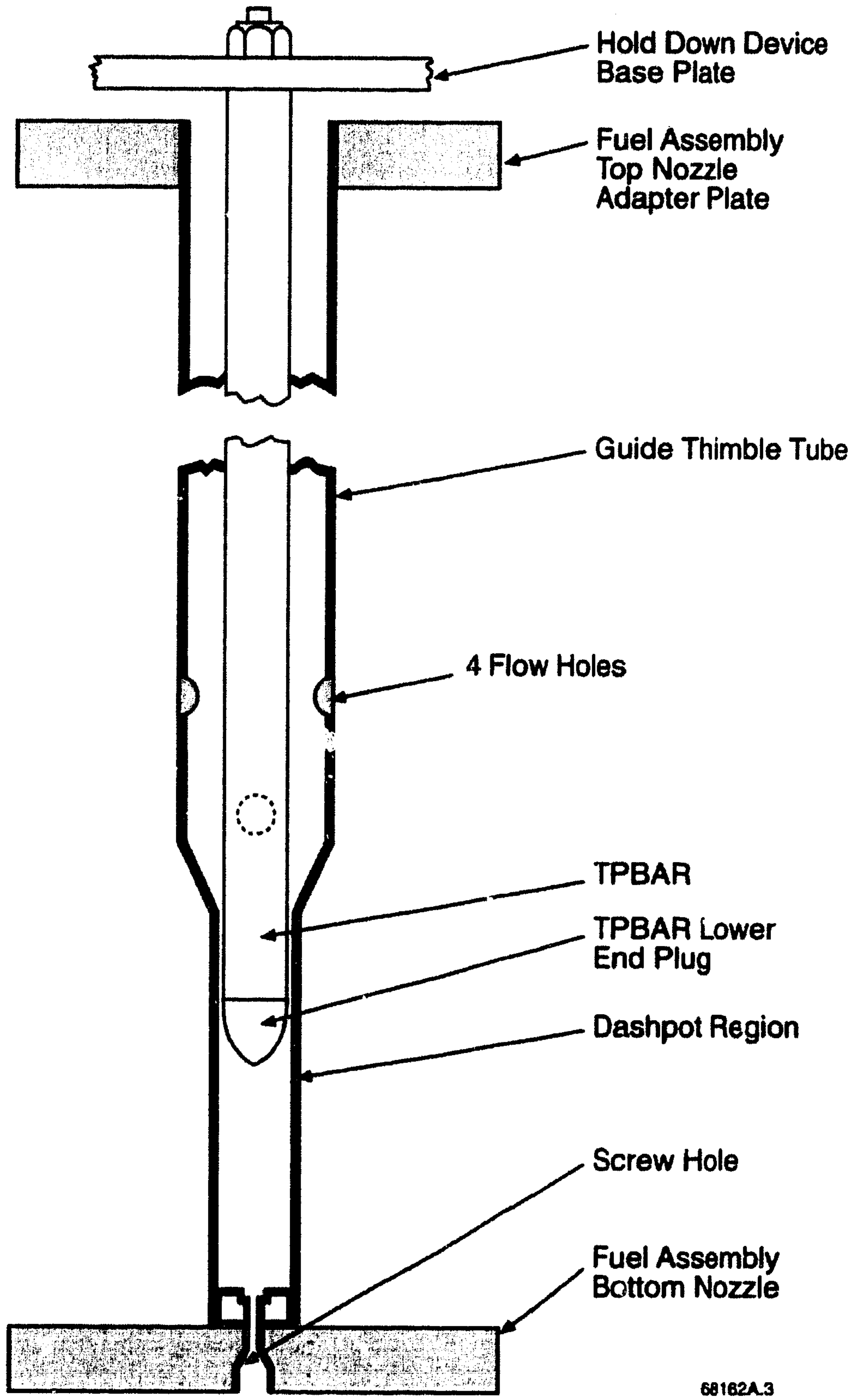

Figure 3.2-4
Fuel Assembly Guide Thimble Tube with TPBAR

3.3 DESIGN REQUIREMENTS

The TPC TPBAR design for the tritium production core shall meet the functional requirements listed in Table 3.3-1. They are essentially identical to the LTA TPBAR requirements. Quantitative performance limits and requirements to be met by the TPBARs in the TPC are listed in Table 3.3-2 and are almost identical to the LTA TPBAR limits, which are shown for comparison. Table 3.3-2 also includes assumptions used in the design of the LTA and TPC TPBARs. These requirements were derived from requirements for commercial core components, such as burnable absorber rods.

The core design described in Section 2.4.3 in conjunction with the thermal hydraulic core conditions, determine the range of TPC TPBAR performance parameters. Table 3.3-3 lists the significant TPBAR parameters used for the average rod, the peak rod, and enveloping generic design conditions.

The TPC TPBAR performance was evaluated for limiting design conditions, the TPC TPBAR in the peak core position, and the average TPC TPBAR. Significant TPC TPBAR generic design characteristics are:

- 1.2 g tritium production in an 18-month cycle for determination of the rod mechanical performance,
- calculated TPC TPBAR peak and average rod tritium production and release values for rod performance evaluations.
- an assumed 1.0 Ci/year tritium release from the core average rod as input to plant evaluation,
- minimum coolant flow in the fuel assembly thimble for thermal evaluations, and
- Thermal Hydraulic performance evaluation performed for a high power density core with 108.04 w/cm^3, and a Westinghouse type 17 X 17 VANTAGE + fuel configuration.

These conditions, in conjunction with other generic assumptions used to evaluate commercial core components, should envelop operating conditions in the majority of PWRs currently operating with equal or less power density.

Application of the design requirements to the TPC TPBAR requires that the function of each feature and component be evaluated for compliance. Table 3.3-4 provides a structure for the evaluation of TPC TPBAR requirements for each component and feature.

The reactor system pressures and temperatures to be used for the TPC TPBAR mechanical evaluation for Conditions I, II, III, and IV transients, referenced in Sections 2.3.2 and 2.15, are listed in Table 3.3-5. These design conditions are typically specified for each plant. Bounding parameters included in this table, however, have been defined in the core component design procedures.

The analysis assumption and design margins calculated were selected so that the TPC TPBARs can be inserted in any core thimble location without the need for additional analysis or evaluation to determine the acceptability of a specific core location.

The impact of the tritium release from TPC TPBARs on other system and environmental releases is determined with a conservative unclassified value. To substantiate that this value is

conservative, analysis is performed for the peak and average TPC TPBAR in the core and compared with calculated values.

The analysis serves also as a means to determine that the TPC TPBAR components, getter liner and pellet perform within their component specific limitations. Since the design production is 1.2 g, the same as for the LTA TPBARs, no change from the evaluation in PNNL-TTQP-6-503, Rev. 1 (Reference 1) is expected. Analysis is performed for the peak and average rod for conservative and nominal conditions to determine the design margins for the TPC TPBAR and components.

For TPC TPBAR designs operating within the limits defined in this topical report, sufficient analyses were performed to verify that the TPBAR in a specific production core design will operate within the constraints defined in the topical report. Table 3.3-6 lists types of evaluations that are required for the TPBARs in a tritium production core.

It is expected that a future partial or full core load with TPBARs in a PWR will contain TPBAR designs comparable to the LTA and TPC TPBAR designs and will operate within the bounds of the limits established and analyzed in this report.

3.3.1 REFERENCES

1. PNNL-TTQP-6-503, Rev 1 (CRD), "Report on the Evaluation of the Tritium Producing Burnable Absorber Rod Lead Test Assembly", February 1997.

2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Department of Energy's proposal for the irradiation of the lead test assemblies containing tritium producing burnable absorber rods in commercial light water reactors", Project No. 697, April 1997.

Table 3.3-1 TPC TPBAR Functional Requirements
1. Structural integrity of the TPC TPBAR shall be maintained throughout Conditions I and II and during shipping and handling.
2. Impact of TPC TPBAR rupture during accident conditions shall be bounded by existing safety analyses limits and offsite/onsite dose shall not exceed 10CFR100 limits.
3. Swelling or shrinking of internal TPC TPBAR components shall be accommodated by the design to ensure removability of the TPBARs from the fuel assembly.
4. The TPC TPBAR cladding stresses and the end plug weld stresses shall not result in cladding collapse, excess ovality, or cracking over the irradiation life of the TPBAR.
5. The cladding shall be free standing and shall not collapse due to external pressure or creep for a design life of 520 EFPD.
6. The TPC TPBAR shall not fail due to vibration fatigue, design cycle fatigue or fretting wear resulting from reactor coolant flow-induced vibration. The host guide thimble shall not fail by fretting wear resulting from reactor coolant flow-induced vibration. The presence of the TPBAR shall not adversely impact the vibration fatigue or design cycle fatigue performance of the host guide thimble.
7. Corrosion and erosion of the TPC TPBAR outer surface shall not cause material transfer into the reactor coolant in excess of rates comparable with other reactor internal components.
8. The absorber pellet structural integrity shall be maintained over the irradiation life of the TPC TPBAR.
9. The plenum spring shall have sufficient preload and spring rate to prevent movement of the pencil column stack during fabrication, shipping and handling, considering a 4g axial acceleration loading at beginning of reactor core life.
10. The TPC TPBAR shall be sufficiently straight to allow insertion into a fuel assembly and shall maintain dimensional integrity to allow removal from an irradiated fuel assembly without excessive force.
11. The TPC TPBAR shall be similar in its nuclear characteristics to a BPRA and be compatible with the nuclear design requirements.
12. The maximum coolant temperature in a guide thimble containing a TPC TPBAR shall not exceed the coolant bulk boiling temperature during Condition I. (Additional design criteria are specified in Table 3.6-1.)

Table 3.3-2 TPBAR Design Requirements and Assumptions***		
Subject Item	**TPC**	**LTA**
Maximum Tritium Production, g/rod	1.2	1.2
Core Power Density, w/cm^3	108.04	103.7
GVR limit, rod average **	215	215
Rod internal pressure limit, psia at 675°F	3200	3000
TPBAR cladding wall temperature limit, °F @2250 psia system pressure	660	660
Max cladding temperature during Conditions I and II, °F	660	650
Bulk boiling temperature in the thimble, °F	~652.7	*
Maximum cladding structural design temperature, °F	660	660
System pressure, psia	2250	2250
System design pressure, psia	2500	2500
TPBAR life-time, EFPD (nominal without margin)	494	495
Mechanical design life-time, EFPD	520	550
Capacity factor, %	90	*
Tritium release per rod average, Ci/year	<1.0	<6.7

* Parameter not evaluated or required for the LTA

** Gas volume ratio based on theroretical density of lithium aluminate

***Use ASME Code stress criteria with Westinghouse generic design stresses for core component rods following the procedure in the Mechanical Design Manual for core rod components.

Table 3.3-3 Significant TPBAR Parameters		
Subject Item	**TPC Data**	**LTA Data**
Number of TPBARs in core	3342/3344 FC/EC	32
Number of TPBAR assemblies	140	4
Number of TPBARs per assembly	24 (most assemblies)	8
<u>TPBAR GEOMETRY & DESIGN</u>		
Cladding OD, in.	0.381	0.381
Cladding ID, in. (before coating)	0.336	0.336
Rod OD tolerance, in.	0.0005	0.0005
Rod length, in.	152.37	152.37
Pellet OD, in.	0.303	0.303
Pellet ID, in.	0.223	0.223
Sensitive information has been removed		
^{6}Li loading, g/in. (enriched pellets)	0.030	0.024
^{6}Li enrichment, % (enriched pellets)	25.3	20.81
Enriched pellet stack length, in., Maximum	127.5 FC/128.5 EC	142
Pellet stack off-set down from centerline, in.	0.50/0.25 FC/EC	0
Rod back-fill pressure, psia	14.7	14.7
Guide thimble OD, in.	0.474	0.474
Guide thimble ID, in.	[ ]^{a,c}	[ ]^{a,c}
Guide thimble dash pot region ID, in.	[ ]^{a,c}	[ ]^{a,c}
FC/EC – First Cycle/Equilibrium Cycle		

Table 3.3-3 Significant TPBAR Parameters (Continued)		
Subject Item	**TPC Data**	**LTA Data**
PERFORMANCE PARAMETERS, TPBAR NUCLEAR INPUT		
Core Power Density, W/cm^3	108.04	103.7
Average fuel rod power, kW/ft	5.68	5.45
TPBAR average rod power, total , kW (with 8 % uncertainty)	5.99	*
Peak rod power, total, kW	8.27	7.83 BOL
Axial average rod power, n/α, kW	4.16	4.53 BOL **
Peak rod power, n/α, kW, includes 4 % inter-assembly gradient factor	5.86	5.4 BOL **
Axial average rod power, n/α, kW/ft, averaged over active pellet stack length	0.391	0.38 BOL
Axial peak of average rod power, n/α, kW/ft	0.466	0.459 BOL
Average rod power, kW/ft with uncertainties	0.498	0.55 (BOL)
Rod average power, γ heat, kW	1.83	1.053 (BOL) **
Peak rod power, γ heat, kW	2.42	1.256 BOL
Intra-assembly power, % (peak)	4	4
Prediction/measurement uncertainty, %	8	5
Total power uncertainty	1.12	1.193

Notes:
1. Heating rates are for steady state operation
2. 8% uncertainties applied to heating rates
3. Upper limit tolerance ^{6}Li loading assumed, 4.2 % tolerance
4. Inter-assembly gradient factor considered is 4 % uncertainty
5. First cycle data unless specified otherwise (bounds equilibrium cycle data)

* Parameter not evaluated/required for the LTA

** Parameter does not contain uncertainty

Table 3.3-3 Significant TPBAR Parameters (Continued)		
Subject Item	**TPC Data**	**LTA Data**
F_Q	2.5	2.5
$F_{\Delta H}$ with uncertainties TPBAR Fuel	1.46 1.65	1.46 1.60
Overpower for Condition II, axial average	1.187	1.21
SURROUNDING FUEL ASSEMBLY DESIGN		
Core average axial peak thermal flux, $n/cm^2/s$,	0.446E14 BOL 0.528E14 EOL	~1.000E13
Axial peak to average neutron flux ratio (Fz)	1.058 BOL 1.112 EOL	1.2
TPBAR Cladding fast neutron flux, >1 MeV, $n/cm^2/s$ in hot assembly (6,1) location, total flux x 0.24	1.06E14 BOL 1.05E14EOL	1.00E14
TRITIUM PRODUCTION IN FIRST CYCLE (FC)/ EQUILIBRIUM CYCLE (EC)		
Tritium production for mechanical and other design assumptions, g	1.2	1.2
Average tritium produced per rod, g	0.856/0.839 FC/EC	0.886
Peak Tritium produced per rod, g includes 4 % factor for inter-assy. gradient	1.089	1.2*
Amount of Tritium produced per cycle, g	2860/2805 FC/EC	28
Rod average GVR	139/137 FC/EC	128
Axial peak GVR in average rod	156/153.8 FC/EC	<153.5
Axial average GVR in peak rod	174	174
Axial peak GVR in peak rod	195	208
Rod average ^6Li burnup, %	45.4/44.2 FC/EC	65
Note: Fluxes given for first cycle are larger than equilibrium cycle fluxes * Without 4% gradient factor		

Table 3.3-3
Significant TPBAR Design Parameters (Continued)

Sensitive information has been removed		

Table 3.3-4
Tritium Production TPBAR Requirements and Evaluation
Sheet 1

TPC TPBAR

- Functional Requirements
- Design Conditions
- Drawings & Specifications
- Nuclear Design, average and peak rod
  - Production/Power
  - Power peaking
- Thermal Hydraulic Evaluation
  - Thimble flow and pressure drop
  - Margin to boiling
  - Cladding temperatures
  - Rod component internal temperatures
  - Operational Impacts of Operating with TPBAR see Sheet 4
- Materials, see Sheet 3
- Mechanical Performance for Design Conditions I, II, III and IV, see Sheet 2

Table 3.3-4

Tritium Production TPBAR Requirements and Evaluation (Continued)

Sheet 2

<u>Mechanical Performance for Design Conditions I, II, III, and IV</u>

- <u>Cladding and End Plug Evaluations</u>
  - Rod Pressures and Temperatures
  - Stresses and Limits
  - Strains
  - Cladding, Buckling, and Ovalization
  - Design Cycle Fatigue
  - Vibration and Wear Damage
  - Cladding Properties and SCC Propensity
- <u>Absorber Pellets</u>
  - Irradiation Stability (GVR Limit)
  - Pellet Chemical Stability
- <u>Getter and Liner</u>
  - Chemical Compatibility
  - Stability of Getter and Liner
  - Chemical Compatibility
- <u>Plenum Spring</u>
  - Material
  - Spring Function
- <u>TPBAR Performance Model</u>
  - Component Temperatures
  - Helium Internal Pressure
  - Tritium Gettering and Partial Pressure
  - Tritium Release
  - Hydrogen Ingress
- <u>TPBAR Performance Model</u>
  - Component Mechanical Interaction (Pellet-Cladding)
  - Rod Axial Growth and F/A Compatibility
  - Rod Bow
  - Component Chemical Compatibility
- <u>Failed TPBAR Performance</u>
  - Rod Failure Mechanism
  - Water Logged Rod

TPBAR/Rod and Component Transient Performance and Stresses

Table 3.3-4
Tritium Production TPBAR Requirements and Evaluation (Continued)
Sheet 3

- Materials
  - Materials Specifications
  - Material Properties
  - SS Materials Performance
    - Type 316 SS Coated Cladding Endplugs
    - Coolant/316 SS Compatibility
    - SCC Propensity
      - Materials and Effect of Coating
      - Stress Condition
      - Coolant and Environment
    - Burst Test Results
    - Welding
    - Materials compatibility (Water Logged Rod)
    - Hydrogen Isotope Permeability
  - Pellet
    - Mechanical Properties
    - Irradiation Performance
  - Getter and Getter Dose
  - Liners
  - Pencils
  - Subassembly Components (spring, getter disc)

Table 3.3-4
Tritium Production TPBAR Requirements and Evaluation (Continued)
Sheet 4

- Operational Impacts of TPBARs
  - Normal Operation
    - Tritium Leakage
  - Refueling Operation
    - Heat Generation
    - Tritium Release Failed Rod
    - Exposure During Refueling
  - Off-Normal Events
    - Absorber Material Relocation
    - Cladding Breach
      - Frequency
      - Defects
      - Breach and Radiological Consequences
    - Misloading a Rod in Improper Position
  - Accident
    - Dropped Rod
    - Anticipated Transient Without Scram
    - LOCA
      - Breach
      - Coolable Geometry
      - Melting, Pellet, Metal
      - Tritium Release
      - Decay Heat Removal

Table 3.3-5
TPC TPBAR Design Bounding Parameters for Conditions I, II, III, and IV

Sensitive information has been removed			

Table 3.3-6
Plant Specific Evaluations Required for TPBARs in PWRs

- Functional Requirements, verify compliance
- Design Conditions, verify compliance with requirements for:
  - Production
  - Cycle length
  - Power density
  - Power peaking
  - Thimble flow
  - Check against generic reactor conditions
- Drawings and Specifications, verify compatibility with assembly design
- Nuclear Design, verify compliance and conservatism of input with limits:
  - Production/Power
  - Power peaking
- Thermal Hydraulic Evaluation, verify conservatism of conditions for:
  - Thimble flow and pressure drop
  - Margin to boiling
- Mechanical performance, verify compliance and conservatism of conditions for:
  - Tritium and Helium production, pressure and cladding stress
  - Pellet GVR limit
  - Getter loading
  - Tritium Release

3.4 MECHANICAL DESIGN EVALUATION

This section provides an evaluation of the TPBAR design for credible combinations of thermal, neutronic, mechanical, and hydraulic interactions. The TPBAR is evaluated for structural integrity of the pressure boundary (cladding and end plugs) and for absorber pellet stability. Evaluations for reaction with and impact on the reactor coolant are described in Section 3.8. Evaluation of the interactions includes the effects of Conditions I, II, III, and IV.

Sensitive information has been removed

The densification and swelling of TPBAR components impact the design tolerances and clearances between component surfaces during operation. Correlations presented in the Materials Property Handbook (MPH) (Reference 6) were used to establish that clearances between components are sufficient to prevent mechanical interactions throughout the life cycle of the TPBARs.

3.4.1 TRITIUM PRODUCTION AND DESIGN LIFE

The TPBAR is currently designed for a maximum production of 1.2g of tritium. The margins between the expected production in TPBAR (see Section 2.4.3), and the design values are shown below.

Tritium Production Design Margin	TPC Design ^3H Production *	Production Design Margin**
Average Tritium Production per Rod, g	0.856FC/0.839EC	1.40FC/1.43EC
Peak Tritium Production per Rod, g	1.089 (FC)	1.10 FC/EC

* See Table 3.3-3
** Relative to 1.2 g

The design life for mechanical evaluation is 520 EFPD. The nominal design life of the core is 494 EFPD. This assumption adds no conservatism to the mechanical design evaluation because the evaluated limited failure modes are generally not time-dependent in the absence of noticeable thermal or radiation induced creep or time dependent material change. The estimated design life can account for tritium releases from TPBARs during partial power operation.

With the 1.2 g limitation and the design lifetime of 494 EFPD, the TPBAR design evaluations show sufficient design margins. The assumptions and design limits applied to the TPC

TPBARs are more conservative than that applied to other commercial core component rods and particularly fuel rods.

3.4.2 CLADDING DESIGN (STRESS, STRAIN, AND STABILITY)

The cladding and end plugs are manufactured from 20% cold worked 316 SS. The cladding is fabricated from seamless tubing coated on the inner surface with an aluminized permeation barrier, and the end plugs are fabricated from bar stock. Credit is taken for the structural benefits of the 20% cold work, with a detractor for recovery of the cold work caused by the barrier coating process. The mechanical properties of the TPBAR cladding are provided in the MPH and in the cladding specification ASTM A 771.

The cladding, end plugs and associated welds form the pressure boundary of the TPBAR. Evaluation of the integrity of the pressure boundary during Conditions I, II, III and IV events is discussed below. The results show that the structural integrity of the TPBAR is acceptable and is maintained during all events under Conditions I through IV (including shipping and handling), with the exception of the large break LOCA events.

3.4.2.1 Stress-Strain

The structural members (cladding and top and bottom end plugs) of the TPBAR were designed using stress and fatigue criteria and methodology consistent with the ASME Boiler & Pressure Vessel Code Section III as a guide. The external pressure criteria of the Code are not applicable because the TPBARs are not reactor core support structure components. Also, strength values used to calculate the TPBAR stresses are based upon material data from the MPH and Reference 1 as the material properties of ASTM A 771 316 SS are not included in the Code. The stress correlation (in Reference 2) is used to evaluate the discontinuity stress at the weld junction between the cladding and end plug (Reference 13).

The loads on TPBARs resulting from worst-case transient pressures, or from handling and shipping, are greater than those due to seismic events. Therefore, operating basis earthquake (OBE) and safe shutdown earthquake (SSE) loads were not evaluated in the cladding stress analysis.

The cladding was analyzed for the most conservative pressure, temperature, and dimensional tolerances for Conditions I, II, III, and IV. For each design condition, the internal design pressure was assumed to be the worst case internal pressure (accounting for non-ideal gas behavior) at the temperature of concern. The limiting stresses for the various stress categories and design conditions are presented in Table 3.4-2. The design stresses were derived from the material properties of 20% cold worked 316SS compiled in the material property handbook, see Section 3.8.2 (Reference 8). The factor of safety to limits and to yield for each design condition is presented in Table 3.4-3.

The results indicate that, except for the large break LOCA event where the TPBARs are assumed to fail, the lowest factor of safety to yield for an in-reactor condition is 1.65, which corresponds to the Loss of Load Without Reactor Trip event (Condition II). Stress analyses of the TPBAR provided the following results:

- Critical buckling pressures were verified to be greater than the RCS design pressure of 2500 psia at the TPBAR design clad temperature of 660°F. The lowest factor of safety based on pressure is 1.60.

- The TPBAR was verified not to collapse or exhibit increased ovality due to the effects of pressure, external temperature, and irradiation-induced creep.
- The TPBAR was verified not to collapse under hydrostatic pressure test conditions (external pressure of 3107 psia at 100°F and 14.7 psia internal pressure), with a factor of safety to yield of 1.71.

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Cold worked 316 SS cladding is stable at the irradiation temperatures and neutron fluence encountered during the in-core residence period for the TPBAR, 660°F and 6×10^{21} n/cm^2 (E>1 MeV). The irradiation creep and volumetric swelling strains are <0.2%. Nominal changes in cladding diametric dimensions due to irradiation creep are plotted in Figure 3.4-1, and are less than 0.0005 in. This is much less than the design limit of 1% on cladding strain.

LBLOCA and SBLOCA Considerations

At the LOCA temperature and pressure listed in Table 3.4-1, the TPBAR cladding stresses exceed design stresses. For abnormal loads during Condition III and IV, ASME code stresses are not considered limiting (Reference 18). Detailed evaluation of cladding failure would need to consider creep deformations which are not addressed.

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Based on this evaluation, a comparison of conservatively calculated rod cladding stresses with measured burst stresses of prototypical cladding, the TPBARs are not expected to fail during a SBLOCA.

Failure of the TPBARs during these events does not interfere with reactor shutdown or emergency cooling of the fuel rods based on the deformation exhibited by cladding in burst tests, see Section 3.8 and Reference 15.

3.4.2.2 Cladding Fatigue

The cladding was evaluated for design cycle fatigue failure due to changes in pressure and temperature during the reactor duty cycle, using the rules of the ASME Code. The cladding satisfies the conditions of NG-3222.4(d), and therefore has the ability to withstand the cyclic service, and an analysis in accordance with NG-3222.4(e) is not required. The design cycle

fatigue evaluation is based on the transient conditions and design cycles for the reference plant, shown in Table 3.4-4.

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3.4.2.3 Cladding Collapse

External pressure tests of barrier-coated cladding are described in Section 3.8. The tests demonstrate that the cladding has adequate strength to resist mechanical buckling from the reactor coolant pressure.

The calculated change in ovality of a TPBAR as a function of time, neutron flux, and uniform external pressure caused by cladding creepdown shows that the TPBAR cladding resists collapse by creep buckling.

3.4.3 ABSORBER PELLETS

The thermal, physical and chemical properties of the absorber pellet are summarized in Section 3.8.

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TPBAR design accommodates the absorber pellet thermal expansion and swelling strain properties as described in the MPH. Lithium aluminate is a high temperature ceramic and is very stable at elevated temperatures. No densification or significant phase change of the absorber pellets is predicted over the range of temperature encountered during Conditions I through IV.

The absorber pellets are chemically stable and are non-reactive with other TPBAR components. The potential for chemical reaction has been evaluated, see Section 3.8. No reaction of lithium aluminate with other TPBAR component materials were found to occur during normal operation or accident conditions (Reference 15).

Experience with irradiation of absorber pellets has shown excellent stability up to a GVR of 239 based on theoretical pellet density (Reference 5), the minimum values achieved in irradiation tests. As discussed in Section 3.9, absorber pellets were irradiated in the ATR to 239 GVR with only minor microcracking. Absorber pellet disintegration, major cracking, and relocation is not expected below the design goal of 215 GVR. As indicated in Table 3.9-5, the maximum calculated GVR is 207 for the TPC. Because the pellets retain their structural integrity, the above design values could eventually be exceeded with additional irradiation exposure.

3.4.3.1 Tritium Loading

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3.4.3.2 Gas Release

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3.4.4 GETTER AND LINER

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3.4.5 PLENUM SPRING

The spring is made from 302 SS and is similar in design to those used in BPRA rods and fuel rods. The spring load stress has been established to be less than 60% of the yield stress, providing a safety factor of 1.66 after consideration has been given to tolerance stackup, internal and external pressure, thermal and radiation growth, compressed height of the spring, and pencil buckling. Based on the safety margin and satisfactory commercial reactor experience with the material in this application, the spring is expected to provide the bearing load required for shipping and handling. No credit is taken for the spring in operational or reactor accident analysis.

Dimensional changes in the plenum spring result from thermal expansion and irradiation growth. These phenomena are described in the MPH.

3.4.6 TPBAR VIBRATION AND WEAR

The requirement is that damaging cladding or thimble wear due to flow-induced vibrations shall not occur.

The potential for damage to occur due to TPBAR vibration was evaluated (Reference 7) by comparing vibration characteristic parameters (frequencies, mass and stiffness) of TPBARs with those of Pyrex BP and WABA rods previously evaluated by Westinghouse. The comparison of the vibration characteristic parameters with BP and WABA characteristic parameters indicated that the TPBARs have similar characteristics and are bounded. In addition, the RMS flow induced vibration amplitude of the three types of rods was calculated consistent with the "S.S.

Chen correlation" in Reference 8. Based on this comparison, the TPBAR vibration amplitudes are smaller than < 0.0035 inch RMS and lower than for the other rods.

BP and WABA rods have been extensively tested and used in PWRs for 30 years. The results of these tests showed that these rods are not prone to flow induced vibration which could result in wear damage of the rod cladding or the thimble tube (References 9 and 10). Experience with BA rods compiled in Reference 19 indicates that rod vibration of BAs confined in a thimble did not cause any component degradation. Therefore, damaging vibration wear of TPBARs in thimble tubes is not expected. In addition, relative to other types of rods, the radial gaps between liner, pellets, getter and cladding in TPBARs should increase the rod internal damping which will reduce vibrations.

The fluid induced TPBAR vibrations generate small cladding bending stresses. The maximum credible vibration stress was calculated to be an alternating stress of 2650 psi, which is bounded by the gap between the TPBAR and the guide thimble. This stress is significantly less than the endurance limit of 24,000 psi specified by the ASME Code. Therefore, failure of a TPBAR due to vibration fatigue is not plausible. BPRA rods used in PWRs have not experienced failure from vibration fatigue.

3.4.7 REFERENCES

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TABLE 3.4-1
TPBAR Design Bounding Parameters for Conditions I, II, III, and IV

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TABLE 3.4-2 Allowable Stress Limits for ASME Code Service Level Conditions				
Design Temperatures °F	Allowable Stress Limits for 20% CW 316 SS Cladding++ Reduced 5% Cold Work Recovery (MPH), ksi*			
Service Levels A, B, and Design				
	S_m	1.5 S_m	3 S_m	4 S_m
200	34.8	52.3	104.5	139.3
384	35.0	52.5	105.0	139.9
618	32.1	48.1	96.1	128.1
650	31.8	47.7	95.4	127.2
660	31.7	47.6	95.3	127.0
683	31.6	47.4	94.8	126.5
Service Levels C, D, and Test Conditions				
	1.5 S_m	2.25 S_m	3 S_m	6 S_m
467	50.7	76.0	101.3	202.6
650	47.7	71.6	95.5	190.9
700	47.3	70.9	94.6	189.1
1447**	14.4	21.6	28.8	57.6
350	52.3	78.4	104.5	209.0
100+	52.3	78.4	104.5	209.0

* Stress limits reported for the LTA are identical for the corresponding temperatures.
** Allowable stress limits not applicable at this temperature.
\+ Hydrostatic pressure test conditions.
++ Derived from 20% CW, 316SS data in the MPH

Table 3.4-3
Factors of Safety (Reference 14)

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TABLE 3.4-4 Transient Conditions for Normal Reactor Operations							
Transient Condition		**Temperature Range**		**Pressure Range**		**Design Cycles**	
		LTA TPBAR	**TPC TPBAR**	**LTA TPBAR**	**TPC TPBAR**	**LTA TPBAR**	**TPC TPBAR**
Heatup/Cooldown @100 °F/h						a, c 15	16
Loading/Unloading 15% to 100%						2175	500
							500
Step Load Increase/Decrease (10% of Full Power)						150	80
							80
Large Step Load Reduction (from 100% to 50% Load)						30	8
Reactor Trip from Full Power						30	17
Hydrostatic Test Pressure						1	1
Leakage Test						30	11
Steady State Fluctuations, Initial, 2 min. Period						N/A	5625
Steady State Fluctuations, Random Fluctuations, 6 min. Period						132.0K	112.5K
Reactor Coolant Pump Startup and Shutdown	Cold					110	120
	Hot						100

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Figure 3.4-1 Effect of Irradiation Creep on TPBAR Cladding Diameter Due to RCS Pressure and Internal Pressure

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Figure 3.4-2 Comparison of SBLOCA Cladding Stresses and Burst Strength

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3.5 TPBAR PERFORMANCE

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The main objective of this section is to describe the basis for the magnitude of tritium losses from the TPBAR to the coolant systems used as input to calculations reported in Section 2.11 of this report. An unclassified tritium release for the average rod of < 1 Ci/rod/year was assumed for the Section 2.11 evaluations.

Calculations were performed for a TPC TPBAR with a design production of 1.2 g of tritium. Table 3.5-1 lists the values for the TPC and, for comparison, the LTA design values. Calculations for the design tritium production are performed to verify that applicable TPBAR component design limits are met. Calculations for the core average TPC TPBAR are also performed to verify that the tritium release to the coolant is less than 1 Ci/rod/year as assumed in analyses performed to support the evaluation in Section 2.11.

Prediction of the tritium loss from a TPBAR requires that the tritium distribution and kinetics in the TPBAR components be modeled. Tritium loss from the TPBAR through the cladding is dependent on the partial pressure of the tritium adjacent to the cladding. An integrated calculation is performed to determine the tritium production in the pellets, the component temperatures, the absorption kinetics of the tritium by the getter and finally the tritium diffusion through the cladding into the reactor coolant.

3.5.1 TPBAR PERFORMANCE MODELING

The analytical modeling of TPBAR component functions encompasses calculating the tritium production, rod powers and component temperatures. With these parameters as input, the helium and tritium release from absorber pellets, the rod helium pressure and tritium partial pressures, reaction of T_2O with the liners, tritium absorption by NPZ getters, and tritium permeation and hydrogen ingress through the barrier cladding are determined.

TPBAR temperatures

Each TPBAR resides in the guide thimble within the fuel assembly and is cooled by reactor coolant that flows up the annulus between the TPBAR and the guide thimble tube. The coolant in the annulus is heated slightly by the TPBAR, but gains more heat from the guide thimble which is heated by gamma radiation and heat transfer from the coolant on the outside of the guide thimble. The coolant temperature in the annulus increases from 557°F (292°C) at the bottom to 625°F (329°C) at the top for the reference plant design.

Heat is generated in the TPBAR from two sources: 1) the $^6Li(n,\alpha)^3H$ reaction in the absorber pellets, which produces one triton, one helium atom, and 4.8 MeV of energy per reaction; and 2) gamma heating in the TPBAR components. The core average linear heat generation rate in the TPBAR due to the $^6Li(n,\alpha)^3H$ reaction is 0.39 kW/ft at BOL. The corresponding core peak TPBAR value is 0.55 kW/ft. The nominal gamma heat generation rate, adjusted for the core

power density, is input from Section 2.4.3 and is added to the n-α heating to obtain the total heat rate.

Both normal operating heat loads and 118.7% overpower (Condition II at BOL) were used as input to determine component temperatures. The resulting component temperatures are input to the subsequent calculations and are listed in Table 3.6-5.

TPBAR internal helium pressure

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TPBAR internal tritium pressure

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Pellet tritium release

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Getter tritium absorption

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The amount of tritium stored in the getter as a function of the axial position is shown in Figure 3.5-2. The tritium released from the pellets is shown in Figure 3.5-1 for the analyzed case. The results of the analysis are also summarized in Table 3.5-1. The estimated tritium partial pressure versus time is shown in Figure 3.5-3 for a TPC TPBAR at design conditions.

Tritium permeation through the cladding

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Hydrogen Ingress from the PWR Coolant

Hydrogen (protium) ingress into the target rods from the coolant needs to be considered because it contributes to the total hydrogen loading in the getters. The RCS coolant contains ~35 cm^3/kg STP of hydrogen in water to reduce corrosion of core components. This corresponds to an effective hydrogen pressure of about 26,000 Pa (3.76 psi).

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Estimated axial distribution of the getter loading in the design TPC TPBAR

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3.5.2 TRITIUM RELEASES

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3.5.3 PERFORMANCE DURING ABNORMAL CONDITIONS

Performance evaluations during abnormal conditions assume that a failed or leaking TPBAR will release all tritium to the reactor coolant system. For two rods, see Section 3.7.3, this amounts to <2.4 g per production core during Conditions I and II. However, the failure of two rods at the same time is unlikely and is considered to be an abnormal event.

During a Condition IV, LBLOCA, the cladding is predicted to burst, and it is assumed that all tritium is released to the reactor coolant system.

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3.5.4 FAILURE LIMITS

Breach of the TPBAR cladding during Conditions I and II is unlikely. However, in the event that a TPBAR is breached and reactor coolant enters the TPBAR, the water is expected to partially dissolve the aluminide barrier, releasing insignificant amounts of Al_xO_y, water soluble $AlCl_y$ and suspended solids. The quantities released from 4000 (bounding number) simultaneously breached TPBARs are shown in Section 3.8 to be within specifications for PWR water chemistry. A breached TPBAR is also assumed to release all tritium to the reactor coolant system. Radiological consequences associated with a postulated breached TPBAR are presented in Sections 2.11 and 2.15.6.

In the event of a sudden temperature transient with a waterlogged TPBAR, the low level of heat generation in the TPBAR would cause pressure changes to be sufficiently slow to allow the internal TPBAR pressure to equalize with the RCS pressure without further cladding damage or ejection of internal material. Also, boiling of the water would not occur because of the low heat generation and the increase in heat transfer caused by the replacement of helium inside the TPBAR with water.

Table 3.5-1

**Comparison of Design
Performance Parameters for the TPC and LTA TPBAR**

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Figure 3.5-1
Cumulative Tritium Releases from Top, Mid-plane, and Bottom Absorber Pellets for a Design TPC TPBAR

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Figure 3.5-2
T/Zr, H/Zr, and (T+H)/Zr Ratios Calculated for Design TPC TPBAR Getters – EOL

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Figure 3.5-3
Tritium Partial Pressure Calculated for a Design TPC TPBAR

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Figure 3.5-4
Tritium Release Rate to Reactor Coolant from a TPC TPBAR

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Figure 3.5-5
Cumulative Tritium Release to the Reactor Coolant from a Design TPC TPBAR

3.6 THERMAL-HYDRAULIC DESIGN EVALUATION

The purpose of this evaluation is to determine the effects of the representative reactor core thermal hydraulic conditions on the function and integrity of the TPBARs. Section 2.4.4 addresses the core thermal hydraulics of the fuel with TPBARs inserted.

3.6.1 EVALUATION PROCEDURE

Standard Westinghouse procedures are applied to evaluate the thermal-hydraulic performance of the bypass flow through the fuel assembly guide thimble tubes and the thermal performance of the TPBARs located in the guide thimble tubes.

The Westinghouse methodology was employed to determine for normal operation (Condition I):

- The bypass flow through the fuel assembly guide thimble tubes
- The coolant temperatures
- TPBAR surface temperatures
- The margin to bulk boiling in the coolant flow
- Absence of surface boiling in the guide thimble dashpot

Given the conservatism inherent in the current procedure and the evaluation assumptions, bulk boiling in the guide thimble tubes is not expected to occur.

Separate calculations were employed to determine the rod component temperatures during Condition II operation. Assuming the coolant to be at a conservative upper bounding saturation temperature, the rod internal component temperatures are well below the melting temperatures or temperatures where short-term enhanced gas release from these components may be expected.

The TPBARs are inserted in guide thimble tubes of the fuel assemblies, replacing conventional burnable absorber rods, or guide thimble tube plugs. The TPBARs are cooled by the flow through the guide thimble tubes, which is considered a bypass flow because it does not contribute to the fuel rod cooling.

Figure 3.6-1 is a representative schematic of the coolant flow to a TPBAR inserted in a guide thimble tube. The TPBAR has the same outside dimensions and features as the Westinghouse PYREX burnable poison rod assembly (BPRA), except for plant specific differences in the required poison length based on cycle energy requirements. Coolant flows from the lower end of the fuel assembly into the guide thimble tubes upward to the outlet of the fuel assembly. A small fraction of the total guide thimble tube flow (~ 8%) enters through an orifice at the bottom of the guide thimble tubes as indicated in Figure 3.6-1. The remainder of the guide thimble tube flow enters through the four-side holes in the thimble tube.

The guide thimble tube coolant flow is heated by the TPBAR generated energy, but predominantly by the heat transferred from the coolant in the surrounding fuel channels through the guide thimble tube wall. The guide thimble tube flow is sufficient to prevent inducing bulk boiling in the guide thimble tubes that could enhance corrosion damage.

Guide Thimble Tube Flow and Temperatures

The flow through guide thimble tubes cooling the TPBARs and other core components is determined by analysis with the methodology, which is used in the Westinghouse Thermal-Hydraulic Design Procedure.

The standard thermal-hydraulic methodology is used to investigate the presence of local or bulk boiling in the guide thimble tubes. The coolant inlet temperatures, core pressure drop, loss coefficients, and heat generation rates are input to the analysis. Then the flow rate through guide thimble tube is calculated such that the total pressure drop matches the core pressure drop. Also, the temperature of the bypass flow is determined from the heat transfer through the guide thimble tubes wall and heat generation of the core component rod. The fluid void fraction and bulk temperature are provided as output from these calculations if boiling occurs.

The methodology also calculates the rod surface temperature and guide thimble flow tube temperatures as input to other evaluations. The analysis procedure manual specifies the input requirements and output data.

To obtain the guide thimble tube flow, the methodology solves the continuity, momentum, and energy equations for a flow system consisting of two parallel flow paths. One path is the flow path inside the guide thimble tube or instrumentation tube, see Figure 3.6-1. The other path is the flow path outside the guide thimble tube or instrumentation tube, i.e., the core side. The problem is greatly simplified since the fluid conditions in one channel (the core side) are assumed to be known from other (THINC) calculations. The problem then reduces to that of flow in a single channel where the boundary conditions are specified.

In a series of iterations, the analysis methodology solves the heat transfer equations to determine the amount of heat which is transferred between the core side coolant as a heat source, the guide thimble tube wall, and guide thimble tube coolant. The heat transfer is a function of the temperature differences between core coolant, guide thimble tube coolant, guide thimble tube, and TPBAR component. The calculation process continues until the guide thimble tube exit pressure converges to the fuel assembly exit pressure.

TPBAR Component Temperatures

The TPBAR component temperatures, for cladding, getters, pellets, and liners are determined with simple heat transfer calculations using the coolant temperature for Condition I and the coolant saturation temperature of 652.7°F for Condition II as a heat transfer boundary condition. The component heat generation rates of gamma heating and (n/α) heating in the pellets at 118.7% power for Condition II with generic enveloping power shapes are used to determine the rod component surface temperatures and gradients. The material heat transfer conductance correlation (from the Materials Handbook, Reference 6), gap conductance, and radiation are incorporated into the calculational procedure.

Criteria and Assumptions

Criteria

The TPC TPBARs are designed to meet criteria consistent with those applied to similar core components, described in Section 2.4, such as burnable absorber rods, source rods, and control

rods. A generic analysis was performed on TPBARs to satisfy the criteria given in Table 3.6-1, which are the Westinghouse criteria for core components. Reload specific analysis is required to verify that the bypass flow criterion assumed for the core fuel analysis is met.

Other criteria normally evaluated as part of the generic core component T/H evaluation procedure, such as rod internal pressure determination, are addressed in Sections 3.4 and 3.5 of this report.

<u>Coolant Flow and Temperature Assumptions</u>

The coolant bulk boiling calculations are performed for the following basic assumptions:

- Thermal core design flow
- Worst-case mechanical TPBAR and guide thimble tubes dimensions and tolerances
- Limiting assembly and TPBAR power generation at the $F_{\Delta H}$ specification limit.

The $F_{\Delta H}$ specification limit considered is $F_{\Delta H} = 1.65$. However, for the hottest fuel rod in an assembly, $F_{\Delta H}$ is only 1.42, and for the four rods surrounding the highest power TPBAR, the average $F_{\Delta H}$ is only ~ 1.28. Therefore, assumptions in these calculations are 29% higher than the best estimate powers for this core design. A still conservative coolant temperature is calculated with $F_{\Delta H} = 1.46$ to determine existing margins to bulk boiling. In addition, calculations were performed for a nominal (average rod) condition with 2% uncertainty, $F_{\Delta H} = 1.02$.

Specific evaluation assumptions used in the TPBAR and guide thimble tube evaluation are listed in Table 3.6-2. The thermal-hydraulic parameters used as input to the TPBAR-guide thimble tube flow evaluation are listed in Table 3.6-3.

Given the conservatism of the input assumption and parameters given above, the evaluation procedure does not require applying additional uncertainties to power, temperature, and pressure which are input at nominal conditions.

3.6.2 RESULTS

TPBARs in the TPC generate 38% higher power than equivalent PYREX burnable absorber rods in the same reactor location, primarily due to the higher (n-α) reaction energy release in ^6Li than in ^{10}B. Since the external features of both types of rods are almost identical, the guide thimble tube coolant flow remains unchanged. The TPBAR meets the requirements of Section 3.6.1 above. The results of the thermal-hydraulic evaluation are listed in Table 3.6-4. The results of the component temperature calculations are listed in Table 3.6-5.

No Bulk Boiling

<u>Requirement</u>: There will be no bulk boiling in the guide thimble tubes.

Table 3.6-4 lists the maximum bulk coolant temperature of 652.3°F, 0.4°F below the saturation temperature of 652.7°F for a conservative $F_{\Delta H}$ of 1.65. With a still conservative value of $F_{\Delta H}$ of 1.46, the maximum coolant temperature is 646.8°F and for the nominal $F_{\Delta H}$ of 1.02, the maximum coolant temperature is 622.6°F. Figure 3.6-3 shows the guide thimble tube coolant and TPBAR surface temperature increase over the length of the TPBAR. The maximum coolant temperature is reached at the top of the TPBAR where the TPBAR end plug gamma heating, shown in the power shape of Figure 3.6-2, contribute to the temperature increase.

The TPBAR heat generation (and contribution from the water inside the guide thimble tube) increases the coolant temperature inside the guide thimble by only 6.2°F. The heat transfer from the adjacent fuel rod channels is a major contributor to the coolant temperature inside the guide thimble.

No Surface Boiling In The Dashpot

Requirement: There will be no surface boiling from the core component rod within the dashpot region of the guide thimble tubes, see Section 2.4.4.4.

Figure 3.6-3 shows the rod surface temperature increase in the dashpot region of the rod between 20 and 25 inch from the bottom of the rod. The rod surface temperatures of ~ 600°F are well below any surface boiling temperatures.

Partial blockage of the flow through the screw hole into the dashpot has been postulated in Reference 5. The only source of heat generation in the bottom section of the rod is gamma heating because the pellet column starts approximately 7 inches above the bottom of the core. Because of the low heating rate in this region, no surface boiling in the dashpot will occur with a partially blocked orifice.

Bypass Flow

Requirement: The sum of the bypass flow through all the different types of guide thimble tubes, core component rods and the instrumentation tubes in the core shall not exceed the limits specified.

The design basis for the core thermal hydraulic design is a core bypass flow of 8.4% of the reactor flow. This high bypass flow fraction assumes no plugging of open guide thimble tubes and instrument tubes. With 3342 or 3344 TPBARs inserted, the flow through these guide thimble tubes is reduced by more than a factor of 3 in each location. As a result, the calculated guide thimble tube flow in TPBAR positions is only 1.15% of the total flow through the core. Because the assumed core bypass flow (8.4%) assumes no flow restriction in thimble tubes, inserting TPBARs can only reduce the bypass flow and the above requirement is met.

A total core bypass flow verification is performed as part of the Westinghouse standard reload evaluation and is required for each cycle to demonstrate that the requirements are met. Particularly, any non-standard combinations of core components would be evaluated for each specific reload application to verify there is no violation of the total core bypass limit.

TPBAR Temperatures

Requirement: The maximum temperature of the TPBAR components shall not exceed the melting temperature of component materials during Condition I and II operation.

Guide thimble inlet and outlet coolant temperatures are used as the boundary conditions with a linear distribution between the top and bottom of the TPBAR. Using this coolant temperature profile and predicted heat inputs from the (n-α) reaction and the gamma heating, rod component temperatures at axial nodes along the TPBAR can be calculated. The nodal component temperatures are then used to predict average gas temperatures at representative burnup steps. Section 3.5 provides a description of the specific methodology used in determining the individual component temperatures.

During Condition I events that result in a reactor trip (scram), heat production within the TPBAR drops significantly following the reduction of neutron flux in the core. Margins with respect to the TPBAR thermal design limits are sufficient to ensure no damage to the TPBAR.

During Condition II events that result in a reactor trip (scram), or during steady-state 118.7% over-power, margins with respect to TPBAR thermal design are sufficient to ensure no damage to the TPBAR. Results from the design case TPC TPBAR calculations indicate that during normal operation the component temperatures are highest at BOL. Table 3.6-5 provides a comparison of the TPC TPBAR maximum wall-averaged internal component temperatures during Condition I and II with the LTA TPBAR results and melting limits.

3.6.3 T&H DESIGN SUMMARY

The thermal and hydraulic design basis of the TPBARs ensure that the TPBAR will meet the requirements of Section 3.6.2 and are below the operating temperature analyzed in the structural analysis of Section 3.4.

Standard analytical methods used in the nuclear industry were used to evaluate conditions such as bulk boiling during Condition I operation to ensure that an adequate safety margin exists in the thermal-hydraulic design relative to the criteria. These criteria are similar to those that apply to the Westinghouse BPRAs.

The analyses concluded that the operation with TPBARs in the reactor core is compatible with TPC design performance capability.

3.6.4 REFERENCES

1. WCAP-7113, "Use of Burnable Poison Rods in Westinghouse Pressurized Water Reactors", October 1967.

2. WCAP-10377-A, "Westinghouse Wet Annular Burnable Absorber Evaluation Report", October 1983.

3. Tennessee Valley Authority (TVA), 1994, 1995, "Final Safety Analysis Report for Watts Bar Nuclear Power Plant", Docket Number 390/391, Chattanooga, Tennessee.

4. PNNL-11419, Rev 1, "Report on the Evaluation of the Tritium Producing Burnable Absorber Rod Lead Test Assembly", March 1997.

5. Letter, Robert E. Martin, NRC, to O. D. Kingsley, TVA, Issuance of Amendment on Tritium Producing Burnable Absorber Rod Lead Test Assemblies (TAC No. M98615) September 15, 1997.

6. TTQP-7-008, Rev. 1, "Materials Properties Handbook for the Tritium Target Qualification Project", April 1997.

Table 3.6-1 T/H Core Component Design Criteria
1. There will be no bulk boiling in the guide thimble tubes. 2. There will be no surface boiling from the core component rod within the dashpot region of the guide thimble tubes. 3. The sum of the bypass flows through all the different types of guide thimble tubes-core component rods and the instrumentation tubes in the core shall not exceed the limits specified for the core. 4. The maximum temperature of the TPBAR components shall not exceed the melting temperature of the component material used during Condition I and II operation. 5. Core component rod and assembly lift forces shall be calculated based on the mechanical design flow and the lift force values will be evaluated for acceptance.

Table 3.6-2
Evaluation Assumptions

<u>Guide Thimble Tubes Flow Evaluation</u>

1. The fuel assembly coolant temperatures are calculated for a core flow rate reduced by 8.4% bypass flow. This high bypass flow rate assumes that the guide thimble tubes are not plugged by TPBARs or other core components. Reducing the core flow maximizes the core coolant temperatures and heat transfer into the guide thimble tubes flow.
2. Fabrication tolerances are used to give the worst case for the analysis being performed.
3. Design tolerances were selected to maximize the guide thimble tube gamma heating.
4. The TPBAR power includes the energy deposited in the water flowing through the guide thimble tubes.
5. The plant is operating at 100% power at 2250 psia, and nominal T_{in} for boiling considerations.
6. For boiling analysis, a long-term, steady-state axial power shape is used, see Figure 3.6-2.
7. The TPBAR rod is operating adjacent to the hot fuel rod channel.
8. The thermal condition of the flow channels surrounding the guide thimble tubes is obtained from a representative THINC code evaluation.
9. Calculations are performed for $F_{\Delta H} = 1.65$ and $F_{\Delta H} = 1.46$ for fuel assemblies with Intermediate Flow Mixer Grids (IFMs).

<u>Material Temperature Evaluation</u>

10. Overpower conditions, that is, 118.7% power, are used for maximum TPBAR component temperature calculations.
11. Temperature dependent values of thermal conductivity and thermal expansion coefficient are used.
12. One-dimensional, steady-state heat conduction analysis is used in material temperature calculation.
13. The plant specific total peaking factor, F_Q, is applied for calculation of maximum material temperature.

Table 3.6-3 TPC Guide Thimble Thermal Hydraulic Parameters	
Parameter	**Value**
Reactor Flow Rate, gpm	374400
Core T_{in}, °F	556.8
Core T_{ave}, °F	593.0
System Pressure, psi Core Bypass Flow for Core Fuel Evaluation, % Fuel Assembly Floor Area, in^2 Numbers of Intermediate Flow Mixer Grids	2250 8.4 40.39 3
Guide Thimble Tube Data	
Inside Diameter, inch	[ ] a,c
Guide Thimble Tube Length, inch	156.0
Dashpot Inside Diameter, inch	[ ] a,c
Dashpot Orifice Diameter, inch	
Dashpot End Plug Flow Hole Length, inch	

Table 3.6-4 TPC Guide Thimble Evaluation Results			
Parameter	**Design Condition**	**Maximum Core Condition**	**Nominal Condition**
Fuel Assembly Power Factor, $F_{\Delta H}$	1.65	1.46 (1.42 + 0.04 uncertainty)	1.02
Absorber Heat Generation, BTU	3748800	3317120	2317440
Fuel Assembly Pressure Drop, psi			a,c
Guide Thimble Tube Pressure Drop, psi			
Rod Pressure Drop, psi			
Rod Hydraulic Lift Force, lb			
Dashpot Flow, lb/hr			
Guide Thimble Tube Flow, lb/hr			
Core Guide Thimble Tube Flow, % of Total			
Max. Guide Thimble Tube Bulk Coolant Temp, °F	652.3	646.8	622.6
Max. Cladding Surface Temp, °F	654.5	648.4	623.9
Limit Bulk Coolant Temperature, °F	652.7	652.7	652.7

Table 3.6-5
TPC TPBAR Maximum Wall Average Temperature Compared with the LTA TPBAR Results and Melting Limits

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a, c

Figure 3.6-1
Thimble-TPBAR Coolant Flows Schematic

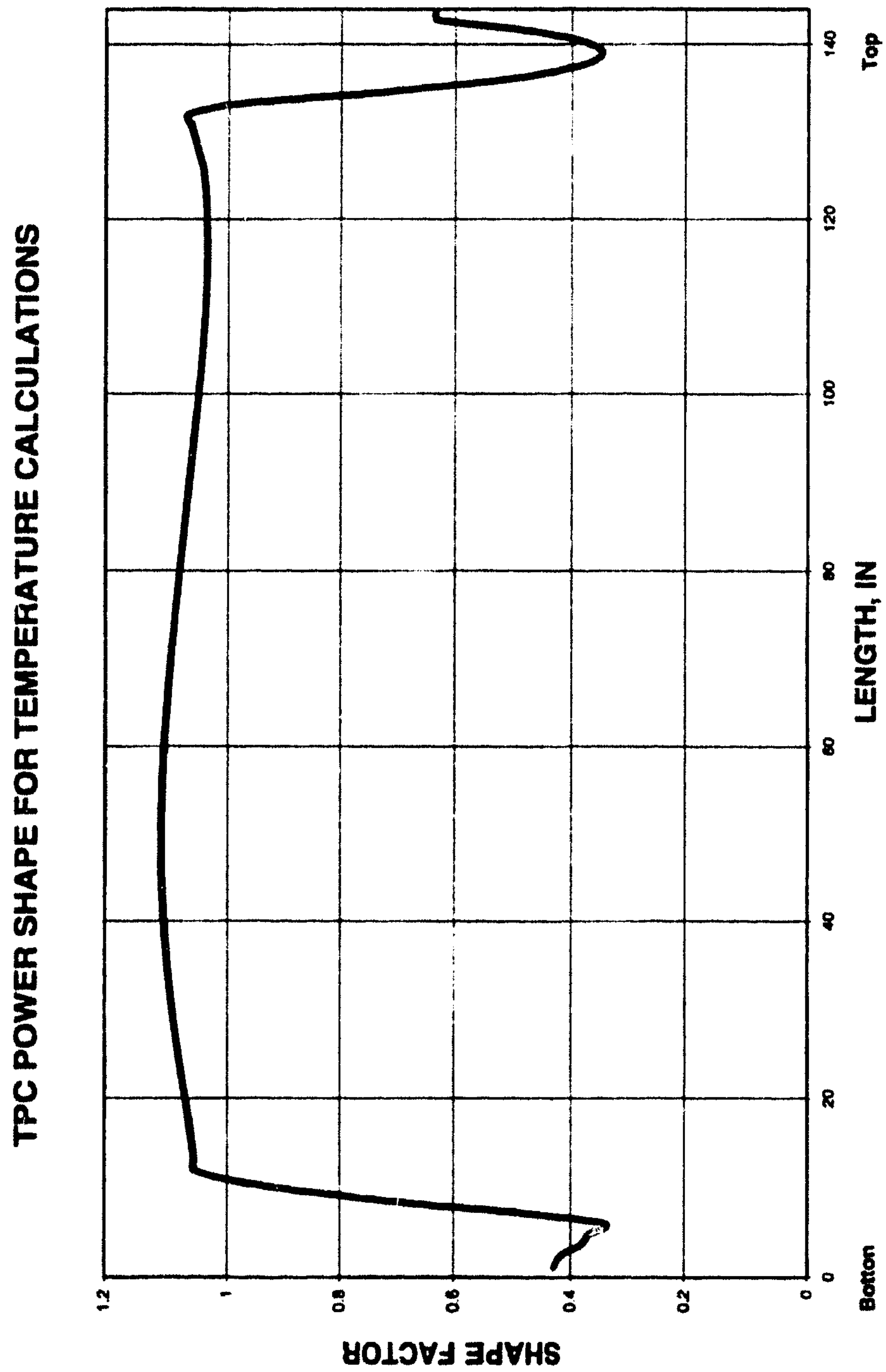

Figure 3.6-2
Power Shape for Temperature Calculations

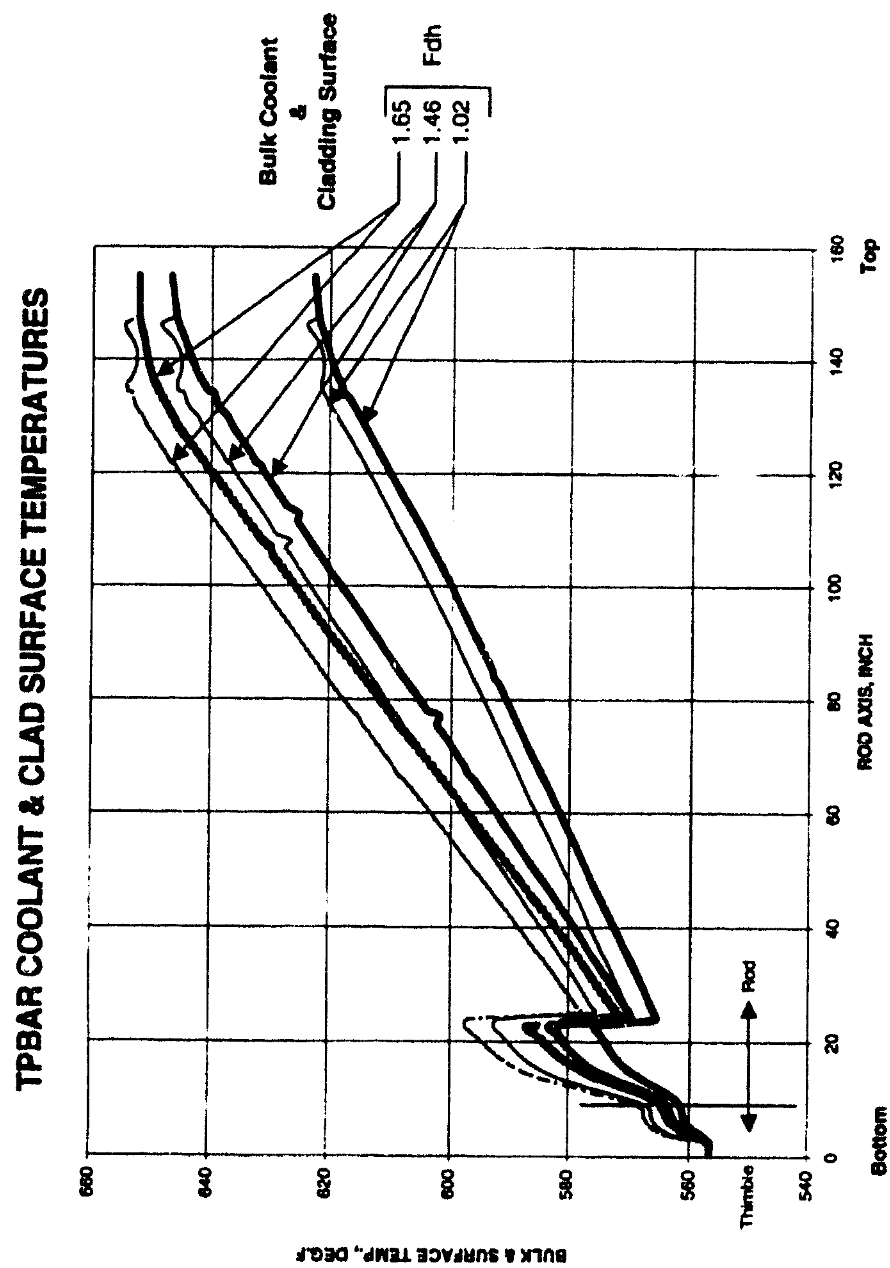

Figure 3.6-3
TPBAR Coolant & Clad Surface Temperatures

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3.7 NUCLEAR DESIGN INTERFACES AND CONDITIONS

3.7.1 LITHIUM-6 PELLET LOADING TOLERANCE REQUIREMENT

The nuclear design defines the required neutron absorbing mass in the TPBARs with a tolerance allowance that maintains the soluble boron concentration in the coolant at a level that assures operation within core operation limits (peaking factors, moderator coefficient, power tilt, boron concentration limits, etc.).

For the reference core design described in Section 2.4, the required lithium-6 loading is 0.030 g/inch, with a tolerance of ±4.2% on the total ^{6}Li loading within any individual pencil. Equivalent requirements have been established for the ^{10}B loading of standard burnable absorber rods. The ^{10}B loading tolerance has been selected for burnable absorber rods for a core wide pellet lot to assure that the critical boron concentration in the reactor coolant of the as-built core will differ from the design calculation by less than ±25 ppm due to variation in burnable absorber ^{10}B loading. In reality and through experience, the ^{10}B loading tolerance is reduced due to randomization during the fabrication process both at the pellet selection and rod selection stage and through randomization of absorber clusters such that ^{10}B loading tolerances in burnable absorbers cause minimal neutronic effects. A similar approach may be used in the fabrication of TPBARs and when loading the core with TPBARs.

<u>Implementation</u>

The higher reactivity worth of the ^{6}Li in the TPC relative to ^{10}B used to control core reactivity, and the current experience base in producing ^{6}Li enriched aluminate, impose a tight ^{6}Li loading tolerance. Analysis of the pellet production data and additional assessment of the ^{6}Li uncertainties result in a ^{6}Li tolerance on an individual pencil basis of 0.030 g/inch ±0.00125 g/inch (i.e., a tolerance of ±4.2 percent). This tolerance appears to be achievable for future pellet production. The pellet sampling plan for the ^{6}Li loading attribute is based on a 95%/95% statistical confidence and acceptance limit.

<u>LTA Pellet Production Experience</u>

The ability of the pellet manufacturer to meet the 4.2% ^{6}Li loading is a function of the fabrication parameter variables and measurement uncertainties. It is necessary to extrapolate from the relatively small production lot manufactured for the LTA to those required for a full core production.

During the manufacture of 32 TPBARs for the LTA, data was obtained on the current production process. The existing statistics are for relatively small lots of pellet stacks (<300 stacks or less than 25 rods).

Two fundamental measurements enter into the ^{6}Li axial distribution (grams of Li-6 per inch): 1) the ^{6}Li assay (in grams of ^{6}Li per grams of pellet) and 2) the pellet axial density (in grams of pellet per inch). The elements that significantly impact the ^{6}Li axial loading tolerance are:

<u>Analytical Chemistry</u>

Uncertainties in analytical chemistry analyses tend to vary with the laboratory performing the work. The analytical laboratory selected to support the TPC pellet production will be required to achieve at least a Relative Standard Deviation (RSD) of 0.80% for ^{6}Li assay measurements. This accuracy was achieved for the LTA pellet

production and is higher than the value of 0.22% achieved by the National Institute of Standards and Technology (NIST).

^{6}Li Standards

Biases in measurements of ^{6}Li are at least 0.9% (References 2 and 3), stemming from issues surrounding the certification of the common lithium standard. Other biases also could be present. An assumption of this evaluation is that these biases will be reduced to insignificant levels prior to commencing pellet production.

Axial ^{6}Li Pellet Distribution

On the basis of LTA TPBAR experience, lot-to-lot variations in ^{6}Li axial mass distribution (grams of ^{6}Li per inch) depend primarily on the pellet manufacture's capability to project powder pressing and sintering outcomes from a test run to a production run. A reasonable assumption for future pellet production is that lot-to-lot variation can be held to ±2.7%. Experience with the LTA pellet production indicates the within-lot variations in pellet production likely will be less than lot-to-lot variations.

TPC Pellet Production

Statistical analysis was performed (Reference 3) to combine the uncertainties affecting the ^{6}Li distribution using statistical procedures acceptable to NIST (Reference 4). In projecting the lithium axial distribution, ^{6}Li/L (^{6}Li gm/inch), from production lots, it is obvious that this parameter is derived from two fundamental measurements. The first measurement is the ^{6}Li assay (^{6}Li/W) in units of ^{6}Li grams per gram of pellet. The second measurement is the pellet axial density (W/L) in units of grams/inch. These two parameters are independent, both absolutely and statistically. Multiplication of these two parameters together provides the lithium axial loading:

$$\frac{^6Li}{L} = \frac{^6Li}{W} * \frac{W}{L}$$

In Table 3.7-1 the specific uncertainty factors anticipated for ^{6}Li/gram measurements are presented. These factors can be used to construct a combined uncertainty.

Uncertainty values were assigned for each attribute in Table 3.7-1 and are shown in Table 3.7-2, (Reference 3). Some of the factors are significant, and others are not.

Insignificant Uncertainty and Bias Factors

- The actual compositional variability of ^{6}Li/gram within the lot of pellet stacks V_{Li} is a minimal contributor to pellet stack variability because of the process used in pellet manufacturing, i.e., two slurry steps, two spray dryer steps, blending, etc.

- Uncertainty ($U_{W/L}$) and bias ($B_{W/L}$) for mensuration and weight of pellets is far less than any of the uncertainties being considered in this evaluation. Considering the measuring equipment used to date, these sources of variability are expected to remain insignificant with respect to the ±4.2% tolerance level.

- The variation of the average ^{6}Li/gram from the manufacturer's intended value for the lot (D_{Li}) can occur, but the impact of the uncertainty is small. Considering the small

changes anticipated in ^{6}Li/Weight during pellet fabrication processes, any deviation of ^{6}Li/Weight can be compensated for by the modifications to the inside diameter, density, and outside diameter. From historical experience, within the bounds of other specification limits, the axial mass loading can be used to compensate for ^{6}Li/Weight deviations between powder lots.

Significant Uncertainty Factors

- U_{Li} is the measurement uncertainty for assay of ^{6}Li/gram in the stack representing the measurements' variation. This value was obtained based on LTA analytical chemistry measurements and averaged for a pencil of pellets. Effects not considered include B_{Li}, the bias in the ^{6}Li measurement. For the purpose of this evaluation it is anticipated the bias in ^{6}Li measurements will be reduced to insignificant levels prior to production.

- $V_{W/L}$ is the variable of axial mass distribution (gram/inch) for the pellet stacks. The axial loading (Weight per Length, W/L) is measured by weighing a stack and measuring its length. In an LTA lot, all pellet stacks were measured and an RSD of 0.41% results for this parameter. The variation of W/L is a product of three parameters: pellet inside diameter, pellet outside diameter, and density. Based on other considerations during the fabrication process, the expected value increases as shown in Table 3.7-2.

- $D_{W/L}$ is the variation of the average axial mass distribution (grams/inch) from the manufacturer's intended value for the lot (i.e., lot-to lot variability). A worst case lot-to-lot deviation of 2.7% in this parameter was considered based on the LTA production experience and the probability of the maximum level to occur in production. It is anticipated that the pellet vendor will aim for a high nominal level of W/L with realization that if a lot initially fails to be accepted the minor cost of regrinding could bring the pellets into specification.

Statistical Prediction of ^{6}Li Loading Variations

By considering the statistical variations in the data from a lot of pellets in the LTA, the stochastic uncertainty components in Table 3.7-2 are combined using NIST and ANSI statistical approaches so that a 3.2% window for lot-to-lot variations in W/L is generated within the overall ±4.2% tolerance level. Expected lot-to-lot variations in grams ^{6}Li/inch are not anticipated to be greater than 2.7% and will, therefore, be acceptable.

3.7.2 ALLOWABLE FUEL PEAKING CAUSED BY AXIAL TPBAR PELLET GAPS

Axial gaps between absorber pellets in a pellet stack or between pellets in adjacent pencils can cause power peaking in adjacent fuel rods.

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The nuclear requirement is that gaps between pellets shall cause power peaking of less than 3% for burnups below 10,000 MWD/MTU and less than 5% for burnups above 10,000 MWD/MTU. There is additional margin to accommodate power peaking in the second half of the cycle because there is a flattening of the core power profile (lower F_{dh}) as the cycle progresses beyond the mid-point. A full-three-dimensional evaluation of the power peaking in the fuel rods adjacent to TPBARs with worst-case gaps demonstrates that this functional requirement will be met.

3.7.3 INTERFACES AND OPERATIONAL IMPACTS

This section addresses the performance of the TPBAR during normal operation, off-normal events and accident conditions and the data and interfaces used in other plant systems that have a functional relationship to TPBARs. The basic TPBAR performance parameters, tritium production and operating environment, are obtained from Sections 2.4, Fuel System Design, Nuclear Design, and T&H Design.

OPERATION (CONDITION I AND II)

Tritium Production

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Tritium Release

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TPBAR Failures

In addition to the expected release from the TPBARs, it is assumed that a maximum of two TPBARs fail or leak after start-up or at some time during the irradiation cycle. No specific failure mechanism is postulated and leaks only due to fabrication defects, wear or handling defects can be envisioned.

Failures of commercial heterogeneous burnable absorber rods (BA) in PWRs have occurred and are reviewed in Reference 5. However, the mechanism for such failures and the concern about such failures are specific to those designs and not applicable to TPBARs.

- BA experience with Al_2O_3-B_4C absorber and Zr-4 cladding:

  Failure mechanism: - Pellets swelling and pellet-clad interaction

  - Zr-4 hydriding
  - Boron leaching from pellets in water logged rods
  - Abnormal rod growth of Zr-4
  - Fabrication defects

  BA rods in some reactors had to be replaced to correct initial design and fabrication problems. The above failure mechanisms (other than fabrication defects) cannot occur in TPBARs due to differences in cladding and absorber materials.

- BA experience with Boro-silicate glass absorber with SS cladding:

  Failure mechanism: - Jammed rods after 3 to 5 cycles (with source assemblies)

  - Unexplained defects (Reference 5)
  - Glass slumping

  In first core irradiation, 2 rods (0.007%) out of 29,700 rods were found to be defective. The reported failures occurred only until initial design problems with glass slumping had been corrected.

There is no evidence of Westinghouse WABA rod failures or deficiencies; large numbers (approximately 500,000) of these rods have been irradiated.

Since the evaluation documented in Reference 5, no additional production or failure statistics for burnable Boro-silicate Glass absorber or WABA rod failures have been compiled. Westinghouse continues to supply WABA and Boro-silicate Glass rods and no malfunction of these rods has been reported after initial startup problems were resolved. The high reliability of this product is consistent with the low incidence of fabrication defects of fuel rods.

This experience is relevant to TPBARs which will be produced to the same high quality standards. Based on the above experience, assuming 2 (0.06%) TPBAR rod failures to occur in one cycle is extremely conservative.

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TPBAR Compatibility with RCS Chemistry

The stainless steel exterior surface of the TPBARs is indistinguishable from other stainless steel components in the core, such as burnable absorbers or fuel assembly end components. The compatibility of stainless with the RCS coolant chemistry is discussed in References 7 and 9.

During normal operation, TPBARs release a minimal amount of tritium to the RCS coolant (<1Ci/TPBAR/Year). Since tritium is chemically indistinguishable from other hydrogen isotopes, tritium in the RCS coolant does not affect the coolant chemistry. The radiological consequences of tritium release to the coolant are addressed in Sections 2.11 and 2.15. Routine monitoring of the tritium in the coolant is required by the RCS chemistry specification and response to detected abnormal tritium activity in the coolant are specified, as for example in Reference 9.

Refueling Operations

The TPBARs will be handled and shipped to the reactor site by methods similar to those applied to burnable absorbers. Prior to shipment to the reactor, the TPBARs are attached to a baseplate, see Figure 3.2-3, and inserted into fuel assemblies at the fuel fabrication facility. Fuel assemblies may be shipped with TPBARs in guide thimble locations in standard shipping containers for fresh fuel, applying standard procedures. Receipt of the TPBAR clusters/fuel assembly combination at the utility will follow the utility's standard receiving, unloading and handling procedures for burnable absorber and fuel assemblies.

Prior to loading into the reactor, the fuel assembly containing the inserted TPBARs will be placed either in the utility's new fuel storage area or in the spent fuel pool. After irradiation, the TPBARs will be stored in the spent fuel pool until they are shipped off-site.

During refueling operation, with refueling and fuel pool temperatures of 140°F to 190°F, the tritium release from TPBARs is very low, much less than 1 Ci/TPBAR/Year and is not considered to affect evaluations. Defective TPBARs moved to the fuel pool could continue to release the stored tritium at a slow rate into the pool. Although no estimate for the release rate is available, the releases are not expected to exceed spent fuel pool limits. The effect of tritium containing TPBARs on the spent fuel pool's radioactivity is addressed in Section 2.12.3.

Following irradiation, each TPBAR assembly will be moved with the host fuel assembly to a storage position. The TPBAR clusters will be stored until they are prepared for shipment by DOE to the extraction facility.

On-Site TPBAR Assembly Movement and Handling

Loading and shipping of the TPBAR clusters will be controlled in accordance with the plant's administrative policies and procedures. The equipment and tools normally used for movement and handling of fuel assemblies and burnable absorbers will be used for the movement and handling of the fuel assemblies with TPBAR clusters. No new or modified tooling will be required for handling of the TPBARs until they are removed from the fuel assembly (which is outside the scope of this report). The external dimensions and materials of the TPBARs are

essentially identical to those of standard core component assemblies, for example with burnable absorber rod assemblies. The plant handling procedure for the burnable poison rod assembly handling is expected to be sufficient for handling TPBARs. For example, a hand winch operated tool is used to remove base plates with attached core component assemblies. There are no provisions for a load-monitoring device to be attached to this tool, which is used to handle BPRAs containing as few as four full-length rods to as many as 24 full-length rods. The TPBAR cluster weight is bounded by the design conditions for handling clusters and no change to the procedure or handling equipment is warranted.

Off-Site Shipping of TPBAR

After removal from the fuel assemblies, the TPBARs will be loaded into a DOE-furnished shipping container suitable for transporting TPBARs containing tritium. The loading and shipment of the TPBARs off-site is outside the scope of this report and will be the responsibility of DOE or an agency assigned by DOE. TPBAR shipment is not a utility responsibility and need not be addressed in the utility documentation and licensing amendments.

One approach for loading and shipping the TPBAR clusters requires a cask outfitted in a manner similar to that which will be used for the LTA shipment. For a larger number of TPBARs, a shipping cask may be outfitted with inserts capable of receiving and holding up to 300 or more TPBARs. A crane will be used to lift the cask in and out of the cask-loading pit. Safe load paths and heights will be controlled in accordance with the reactor plant's procedures.

TPBAR Absorber Material Relocation

The absorber pellets have retained their shape and integrity during previous test irradiation (see Section 3.8 and 3.9). No densification or phase changes of the absorber ceramic over the temperature range of the operating conditions was previously observed and is not expected to occur. Relocation of the pellet material in the pencil, see Figure 3.2-1, is prevented by the capture of the pellets between the getter tube on the outside and a liner in the tube center.

Even though pellet stack movement has not been observed and is prevented by design, a small gap between pellets and the resulting effect on power peaking in surrounding fuel rods is considered and addressed in Section 3.7.2. The effect was shown to be benign for credible gaps.

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The components inside the cladding, cladding coating, NPZ getter, pellet and liner are compatible with each other based on a chemical compatibility evaluation (Reference 12).

Reactions and reaction rates of materials to form eutectics are significant only at maximum LBLOCA temperatures >1800°F.

Radiological consequences of a TPBAR leak are considered in Section 2.11 and 2.12. At the end of the irradiation cycle, each TPBAR is designed to contain 1.2g of tritium, or approximately 11,596 Ci. Any tritium released from a failed TPBAR in the reactor under normal operating conditions will be captured in the reactor coolant and most of it released to the environment over the cycle as a water discharge.

<u>Potential for RCS Coolant Interaction with a Water-Logged Rod</u>

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Sensitive information has been removed

ACCIDENTS

Loss of Coolant Events

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Other Accidents

During other design accidents, Condition III and IV events listed in Table 3.3-5, the TPBARs do not exceed design limits, see Table 3.4-3. Short duration increases of coolant temperatures to

less than 1000 °C (1832°F) during these events cause insignificant tritium release and can be neglected.

<u>Handling Damage of TPBARs</u>

TPBARs could be potentially damaged during handling or insertion into or removal from the host fuel assembly. Of concern are only handling incidents with irradiated TPBARs. Incidents with unirradiated TPBARs can be resolved through normal procedures.

- A TPBAR cluster can be dropped during handling.
- A TPBAR can be bent during handling.

Irradiated TPBAR clusters will only be handled and stored as other irradiated BPRAs. The degree of shielding provided by the water or other material in those areas where the TPBAR clusters will be moved or stored is the same as for spent fuel assemblies. The established minimum water depth above the spent fuel ensures adequate radiation shielding.

The TPBAR external features are the same as for a BPRA. The TPBAR cluster has approximately the same weight as other core component assemblies and is not handled at any greater heights than other fuel inserts (Reference 7). The structural adequacy of the cluster of TPBARs hanging on a baseplate was analyzed and meets all structural requirements, see Section 2.4.2.1, for normal operation handling loads.

The drop of a fuel assembly containing a TPBAR cluster or the drop of a TPBAR cluster in the spent fuel pool is bounded by the fuel assembly load requirements.

If an irradiated TPBAR is breached as a result of mishandling in the spent fuel pool, only a small fraction of the tritium inventory is released. The tritium in the open pores of the pellet (tens of Ci) will be released when water comes in contact with the pellet (Reference 7). Further release may occur gradually due to the limited leaching of the pellets. A tritium release through the breach of a TPBAR by diffusion will provide adequate time to isolate the damaged TPBAR cluster to prevent further release into the pool (see Section 2.15.6.3).

3.7.4 REFERENCES

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Table 3.7-1

Uncertainty Factor for Axial Pellet Density and ^{6}Li Assay

Measurements Uncertainty/Inherent

	Measurement Uncertainty	Measurement Bias	Parameter Variability	Parameter Deviation
^{6}Li/W	U_{Li}	B_{Li}	V_{Li}	D_{Li}
W/L	$U_{W/L}$	$B_{W/L}$	$V_{W/L}$	$D_{W/L}$

U_{Li} is the measurement uncertainty for the assay of ^{6}Li /gram in the pellet stack and represents the variation of measurements derived solely from the measurement process.

$U_{W/L}$ is the measurement uncertainty associated with the weight and length measurements.

B_{Li} is the bias associated with the ^{6}Li /gram measurement.

$B_{W/L}$ is the bias associated with length and weight measurements.

V_{Li} is the compositional variability of ^{6}Li /gram within a lot of pellet stacks.

$V_{W/L}$ is the variability of the axial loading (grams/inch) for a pellet stack.

D_{Li} is variation of average ^{6}Li /gram from the manufacturer's intended value for the lot.

$D_{W/L}$ is the variation of the average axial mass loading (grams/inch) from the manufacturer's intended value for the lot (i.e., lot-to-lot variability).

Table 3.7-2 Projected Uncertainty Factor for Axial Density and ^{6}Li Assay (Two RSD on Per Stack Basis)				
	Measurement Uncertainty U %	Measurement Bias B %	Within Lot Parameter Variability, V (%)	Lot-to-Lot Parameter Deviation, D (%)
^{6}Li/W	0.53	0	0	0
W/L	0	0	0.83	2.7

^{6}Li/W = ^{6}Li by weight
W/L = Weight/Length, Axial Density
RSD = Relative Standard Deviation

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3.8 MATERIALS EVALUATION

The properties of the selected TPBAR materials ensure that the design meets the functional requirements, and that the materials are compatible with other reactor components, plant coolant fluid, and spent fuel pool equipment. The materials have been defined by specifications, material properties needed for design, and tests of their performance. This chapter references the material specifications, material properties, and examinations of the materials.

TPBAR materials are compatible within the range of RCS operating conditions and perform satisfactorily during irradiation in the tritium production core. The RCS chemistry limits are not challenged even in the unlikely event that during operation TPBAR internals are exposed to reactor coolant. Material selections have taken into account the consequences of accidents previously evaluated.

Since the submittal of the LTA TPBAR Evaluation Report and the issuance of the SER, the 32 LTA TPBARs have been fabricated and inserted into the Watts Bar Unit 1 reactor. The same materials used in the LTA are used for the TPC TPBARs. In the safety evaluation of the LTA Technical Report and the license amendment related to the LTA TPBARs insertion into the Watts Bar Nuclear Plant, the NRC concluded that the materials engineering issues of the LTA have been acceptably addressed. In this section, material issues related to incorporating a large number of TPBARs in the TPC are addressed.

3.8.1 MATERIAL SPECIFICATION

The TPBAR materials specifications provide design characteristics that are suitable for the production and retention of tritium during irradiation in a PWR. Earlier tritium target rod designs were irradiated in the Advanced Test Reactor (ATR). Based on experience with those designs, material requirements for tritium producing rods have continued to evolve. Quality standards for material selection, fabrication, handling, storage, and inspection are specified to ensure that important functions are maintained.

The TPBARs are constructed of materials that have been chosen for their ability to perform successfully based on results from commercial experience, in-reactor and ex-reactor test programs, and for their compatibility with the other reactor internals, fuel assemblies, the reactor coolant system, fuel pool equipment and fuel pool cooling systems. The TPBARs are attached to a base plate, a standard core component used to support other components such as BPRAs, and is, therefore, not unique to TPBARs and not included as a TPBAR specification.

Commercial ASTM standards are used for procuring and fabricating the 316 SS cladding and end plug pressure boundary, the Zircaloy-4 liner and getter, nickel plating of getters, and the plenum spring. The applicable standards are summarized in Table 3.8-1 and cited in Reference 1 through Reference 7.

Nondestructive examination of tubular products, fittings, and assemblies ensure that the critical characteristics of the material meet specified acceptance criteria. TPBAR materials and components are subjected to quality inspections to verify compliance with specifications. Inspection methods range from standard visual inspections to discrete and unique methodologies specific to particular characteristics of a component. Where applicable, inspections, tests, and analyses are performed to commercial standards. Certified Material Test

Reports (CMTRs) are provided for all TPBAR components. Operations for which commercial standards are not available, such as unique machining and welding processes, application of aluminide coating, production of lithium aluminate absorber pellets, and assembly of rod components, are performed to specifications developed as part of the fabrication development effort for TPBARs.

The nondestructive examination techniques and applicable standards used during TPBAR fabrication are identified in Table 3.8-2.

The TPBAR cladding and end plugs form the pressure boundary between the TPBAR internal components and the reactor coolant system. The principal methods employed to examine the TPBAR cladding and end plugs are ultrasonic, eddy current, radiography, and helium leak testing. Testing of the cladding and end plugs is conducted in conformance with the codes and standards listed in Table 3.8-2.

After application of the barrier coating, nondestructive examinations are performed to verify the acceptability of the barrier coating in terms of key parameters such as the coating thickness, coating integrity, and coating consistency.

Contamination of the TPBARs is minimized during assembly, and testing is performed prior to shipment to confirm that the specified cleanliness requirements are met.

3.8.2 MATERIAL PROPERTIES

The properties of the TPBAR materials are compiled in a Material Properties Handbook (MPH) (Reference 8). The sources for the materials are provided in the MPH. Conservative design properties are derived from the MPH data and used in the TPBAR evaluations. A matrix of the properties contained in the MPH is summarized in Table 3.8-3. In addition to the matrix in Table 3.8-3, the handbook also addresses liquidus temperatures for binary metal phases and provides a unit conversion table. The 14th Edition of the Chart of the Nuclides published by the General Electric Corporation is cited for applicable data such as atomic weights and Avogadro's Number.

3.8.2.1 Materials Properties During Plant Conditions I and II

Material Properties for TPBAR components are strongly dependent on temperature and time. Therefore, the properties are separately addressed for the plant operating conditions. The Condition I and II events and their impact on the TPBAR components are summarized in Table 3.8-4.

<u>Cladding Tritium Permeation Barrier</u>

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Design Strength and Fatigue

The 20% cold worked ASTM A 771 316 SS cladding has a much higher allowable stress (References 1 and 8) between 100 and 850°F when compared to the allowable stress for annealed 304 SS (Reference 9). A comparison is shown in Figure 3.8-1. The fatigue endurance limit provided by Table I-9.2.1 in Reference 9 is 24,000 psi for 1×10^{11} cycles. This corresponds to a frequency of 2226 Hz over the design life of 520 EFPD.

Effect of Barrier Application

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Cladding Collapse

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Rod Internal Components

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Sensitive information has been removed

3.8.2.2 Materials Properties During SBLOCA and LBLOCA

Based on the temperatures provided in the MPH, even during a LBLOCA, the TPBAR component peak temperature is below their respective melting temperatures for the pellet, cladding and zirconium.

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Sensitive information has been removed

Sensitive information has been removed

3.8.3 MATERIAL COMPATIBILITIES FOR NORMAL AND ACCIDENT CONDITIONS

TPBAR 316 SS cladding is resistant to chemical attack from the chemical species normally present in the reactor coolant during Conditions I, II, and III. As presented below and summarized in Table 3.8-4, metallurgical reactions of melting and rapid oxidation result only during Condition IV, but are limited by the short duration.

3.8.3.1 Materials Compatibilities During Plant Conditions I, II, and III

As summarized in Table 3.8-4 and discussed below, the low temperatures, stresses, and low oxidation potential of the coolant during Condition I and II events and the short duration of Condition III events result in negligible cladding corrosion and component interactions.

Cladding Corrosion

Experience in operating PWR and BWR plants with stainless steel clad fuel rods, control rods, and structural components (e.g., end plate castings and support grids) confirms that uniform corrosion of Type 304 stainless steel (304 SS) is negligible, <0.0001 in./y (Reference 16). Compared to 304 SS, 316 SS exhibits better general corrosion resistance (Reference 17); better resistance to pitting corrosion (Reference 18); higher resistance to transgranular stress corrosion cracking (TGSCC) (Reference 19) and to intergranular stress corrosion cracking (IGSCC) in aggressive environments (Reference 20); and greater strength and resistance to creep. Because 316 SS has not been extensively used for in-reactor corrosion studies, data for the uniform corrosion of 304 SS were used to estimate the cladding wastage from corrosion by the reactor coolant. Based on the corrosion rates for 304 SS in PWRs (Reference 16) and in the Engineering Test Reactor, it was estimated that the corrosion rate for TPBAR 316 SS cladding and end plugs is <0.0001 in./y. The design value for cladding wastage in a CLWR due to uniform corrosion of the TPBAR external surface during 520 effective full power days (EFPD) is <0.0003 in. (Reference 8).

Stress corrosion cracking (SCC) in Series 300 stainless steel requires sensitization, an aggressive environment, and high stresses; and may be aggravated by high neutron fluence, hydrogen, and high temperatures. The formation of oxidizing species is effectively suppressed in PWR coolant (Reference 16). Austenitic SS is not susceptible to SCC in PWR coolant, because of the low oxygen concentration (<100 ppb). Cladding is not susceptible to SCC during Condition III events because the cladding tensile stress is low and the duration is short.

Experience with SS-clad fuel and BPRAs in PWRs indicates that with the current PWR water chemistry, crud deposition is acceptably low (Reference 21). TPBARs are designed to be free of

crevices and therefore crevice corrosion is not of concern. Experience has shown erosion of austenitic SS clad BPRA rods to be insignificant. The wear resistance of 316 SS further ensures that the erosion of the TPBARs will be acceptably small.

Compatibility of the TPBAR Internal and Cladding Materials

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Cladding Defects

Leaking of the TPBAR cladding during Conditions I and II is unlikely. Nevertheless, safety analysis criteria assume as many as two leaking TPBARs per production core during Conditions I and II. In the event that a TPBAR develops a leak, reactor coolant water may enter the TPBAR and potentially dissolve some of the aluminide barrier, releasing insignificant amounts of Al_2O_y, water soluble $AlCl_y$ and suspended solids. The quantities released from 4000 (bounding number) simultaneously breached TPBARs are shown in Table 3.8-5 to be within specifications for PWR water chemistry.

In the event of a sudden temperature transient with a waterlogged TPBAR, the low level of heat generation in the TPBAR would cause pressure changes to be sufficiently slow to allow the internal TPBAR pressure to equalize with the RCS pressure. Also, boiling of the water would not occur because of the low heat generation and the increase in heat transfer with water inside the TPBAR. Radiological consequences associated with a postulated breached TPBAR are presented in Sections 2.12 and 2.15.

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Lithium aluminate absorber pellets dissolve acceptable amounts of lithium and aluminum and do not disintegrate in water (References 23 and 33). Lithium aluminate pellets may have limited leaching of lithium in water. Because of the high density and stability of the absorber pellets and confinement by getter and liner tubes within the cladding, significant loss of material is extremely remote. Therefore, in the event of a cladding breach, absorber pellets remain intact and ^6Li is not redistributed.

The 302 SS plenum spring is nonreactive with the other TPBAR components. The spring is only slightly soluble in the reactor coolant. In the event of a cladding breach, a very small quantity of SS will dissolve in the reactor coolant.

3.8.3.2 Materials Compatibility During Plant Condition IV

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3.8.4 REFERENCES

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Table 3.8-1

TPBAR Materials and Assembly Specifications

COMPONENT	MATERIAL SPECIFICATION
PRESSURE BOUNDARY	
316 SS Bar Stock	ASTM A 831-95 (Reference 2)
316 SS Top and Bottom End Plugs	Internal Specification (Reference 28)
316 SS Seamless Cladding Tubes	ASTM A 771-88 (Reference 1)
Aluminized Cladding Inner Surface	Internal Specification (Reference 29)
ABSORBER PELLETS	
Enriched Annular $LiAlO_2$ Pellets	Internal Specification (Reference 30)
GETTER TUBES AND DISCS	
Zircaloy-4 Getter Tubes	ASTM B 353-91 (Reference 3) Grade UNS R60804
Zircaloy-4 Getter Discs	ASTM B 352-92 (Reference 4) Grade R60804
Nickel Plating	ASTM B 689-90 (Reference 5)
LINERS	
Liner Tubes	ASTM B 353-91 (Reference 3) Grade UNS R60804
SPRING	
Plenum Spring	ASTM A 313-95 (Reference 6)

Table 3.8-2

Nondestructive Testing Techniques and Applicable Standards for Acceptance of TPBAR and TPBAR Components

TPBAR Component	Method	Applicable Standard	Characteristic
Coated Cladding	Eddy Current	Internal Specification (References 29, 31)	Coating thickness, uniformity of thickness along the tube, and inter-metallic phase
Final TPBAR Assembly	Radiography	NE F3-10*	Welds, component placement
Final TPBAR Assembly	Helium Leak Test	ASME B&PV, Section V, Article 10	Rod cladding integrity (leak tightness)
316 SS Bar Stock	Ultrasonic	ASTM E 213-93	Defects
316 SS Bar Stock	Liquid Penetrant	ASTM E 165-95	End defects
Cladding tubing	Ultrasonic	ASTM E 213-93	Wall thickness, flaws
Plated getters (tubes and discs)	X-ray spectrometry	ASTM B 568-91	Plating thickness and coverage

* DOE specification developed for fast breeder reactor program.

Table 3.8-3

Properties of TPBAR Materials (Reference 8)

(Table entries refer to section numbers in Materials Properties Handbook, TTQP-7-008.)

Material/ Property	316 SS CW/A*	302 SS	He	Ni	Zircaloy-4	$LiAlO_2$	Barrier Coating	^3H
Composition	SS-0	SS302-0						
Density	SS-1	SS302-1		NI-1	ZR-1	LI-1		
Thermal Conductivity	SS-2		HE-2		ZR-2	LI-2		
Thermal Expansion	SS-3			NI-3	ZR-3	LI-3		
Melting Point	SS-4			NI-4	ZR-4	LI-4		
Specific Heat	SS-5				ZR-5	LI-5		
Yield Strength	SS-6				ZR-6			
Young's Modulus	SS-7				ZR-7			
Poisson's Ratio	SS-8							
Burst Pressure	SS-9							
Corrosion Rate	SS-10	SS302-10					B-10	
Swelling					ZR-11	LI-11		
Irradiation-Induced Deformation	SS-12							
Tritium Permeability	SS-13						B-14	
Ultimate Tensile Strength	SS-14	SS302-14						
Thermal Accommodation (Jump Distance)			HE-15					
Helium Release Rate						LI-16		
Tritium Release Rate						LI-17		
Gettering Rate					ZR-18			
Hydrogen Ingress							B-19	

Table 3.8-3 (continued) Properties of TPBAR Materials (Reference 8)								
Material/ Property	**316 SS CW/A***	**302 SS**	**He**	**Ni**	**Zircaloy-4**	**$LiAlO_2$**	**Barrier Coating**	**^3H**
Axial Growth					ZR-20			
Van der Waals Constants			HE-21					
Half-Life and Decay Constant								T-22
Curie Conversion								T-23
Energy of Production Reaction								T-24
Strain Limit	SS-25							
Torsional Shear Modulus		SS302-26						

* CW = 20% Cold Worked; A = Annealed

Table 3.8-4

Effect of Most Severe Plant Condition on TPBAR Components

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Table 3.8-5 Effect of Cladding Breach on RCS Impurities		
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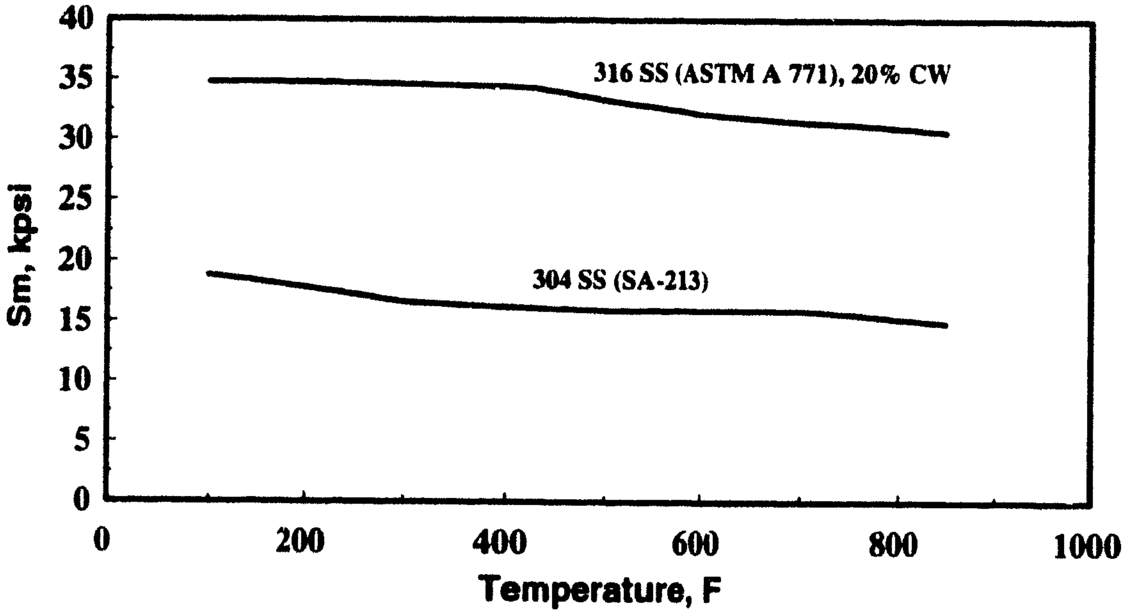

Figure 3.8-1
Comparison of Sm (ASME Code Allowable Stress) for 20% CW ASTM A 771 316 SS (References 1 and 8) with SA-213 304 SS (Reference 9)

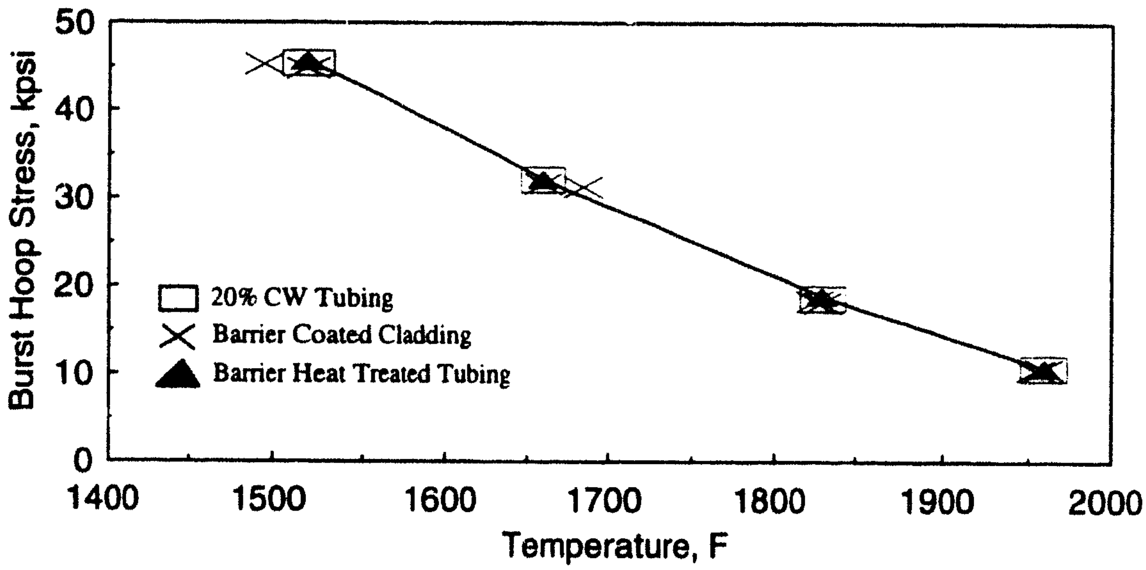

Figure 3.8-2
Effect of Temperature on Cladding Burst for 20% CW 316 SS Tubing, Barrier-Coated Cladding, and Cladding Heat Treated to Simulate the Coating Process

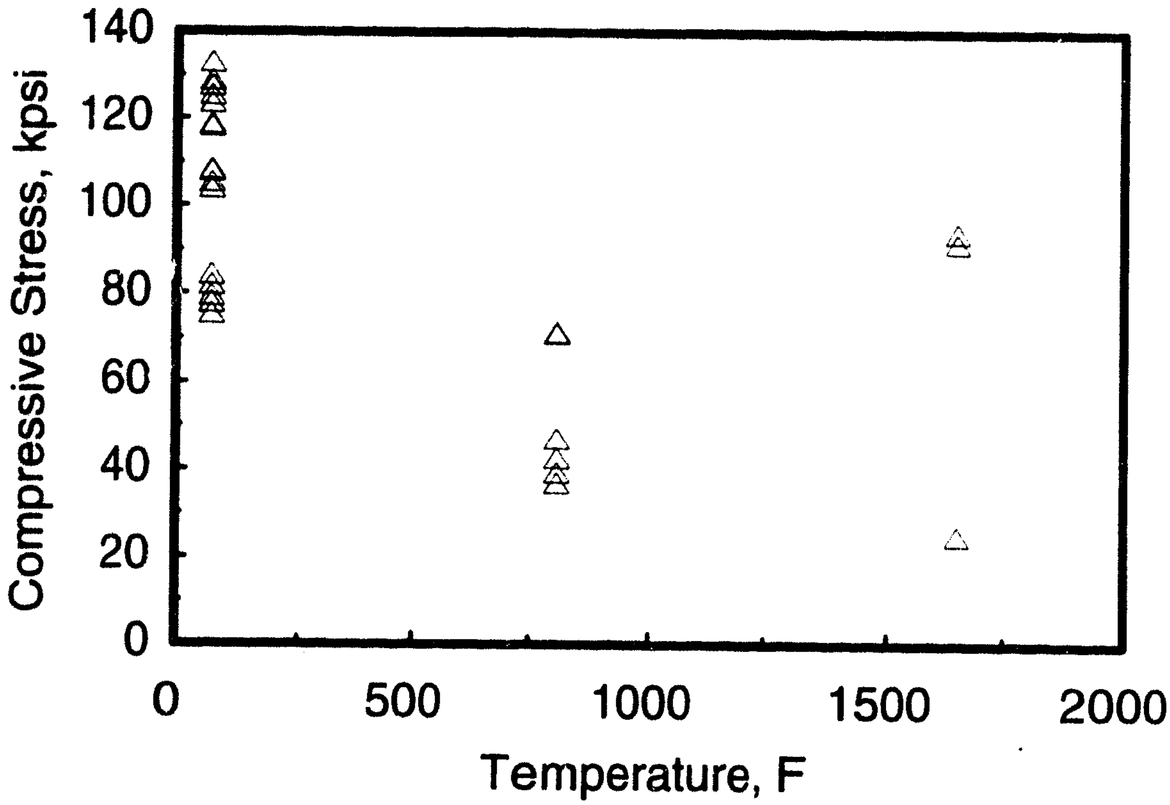

Figure 3.£-3
Compressive Stress to Crush 0.040-in.-Wall Annular Absorber Pellets

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Figure 3.8-4

3.9 TPBAR IRRADIATION TESTS AND TEST DATA SUMMARY

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TABLE 3.9-1

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TABLE 3.9-2

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TABLE 3.9-3

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TABLE 3.9-4				
	Sensitive information has been removed			

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Figure 3.9-1

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Figure 3.9-2

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3.10 PLANNED POST-IRRADIATION EXAMINATIONS FOR THE LTA TPBAR

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The 32 LTA TPBARs began operation in the Watts Bar Unit 1 Cycle 2 in October 1997. No special instrumentation was required or requested for monitoring the TPBAR performance. Hence, the estimates of TPBAR operating conditions are based upon the known general operating conditions of Watts Bar and the methodology developed for prediction of irradiation performance, pressures, temperatures, and tritium production. Tritium loss from the LTA TPBARs cannot be specifically measured in-reactor due to the tritium in the reactor coolant from other sources. The tritium measurements are being evaluated in a comparison with pre-LTA (Cycle-1) Watts Bar data

After irradiation, the LTA TPBAR performance will be compared with the data obtained from nondestructive (NDE) and destructive post-irradiation examinations (PIE), as it was done on the ATR test rods. The planned PIE will be responsive to information that corroborates the general functional requirements for the production TPBAR design and of the information that can be used to advance the analytical models and assumptions.

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Logistics for the PIE are described below in Section 3.10.1. The NDE to be conducted on the TPBARs are described in Section 3.10.2. Special tests and destructive examinations are planned on four TPBARs to provide detailed information related to performance modeling (Section 3.10.3). Finally, the applicability of the examinations for verifying the functional requirements is discussed in Section 3.10.4. It should be noted that most of the TPBAR functional requirements would be addressed by NDE (including rod puncture) which will be conducted on all 32 LTA TPBARs.

3.10.1 CURRENTLY PLANNED PIE STRATEGY AND LOGISTICS

Following irradiation in the Watts Bar Nuclear Plant, all 32 TPBARs will be subjected to NDE. There may be some variation in the intensity of these examinations depending on the initial findings, but in general, every TPBAR will be subjected to visual examination and photography, gamma scanning, neutron radiography, diameter profilometry, and rod puncture/gas analysis, as described in Section 3.10.2.

In addition to the NDE, selected rods will be sectioned for destructive and quantitative examinations and testing. Currently, it is expected that four rods will be selected for destructive examination beyond normal characterization of pre-irradiated components.

3.10.2 NONDESTRUCTIVE EXAMINATIONS

It is planned that each of the 32 TPBARs will be subjected to the following suite of "nondestructive" examinations:

3.10.2.1 Visual Examination and Photography

Rods will be examined visually over the full length in at least two orthogonal orientations, and observations will be documented by photo or video. Unusual features will be noted and recorded; these may include excessive wear, scratches, pits, dimples, patterned corrosion layers, etc.

3.10.2.2 Rod Diameter and Length Measurement

The axial distribution of rod diameter, and the rod tip-to-tip length, will be measured.

3.10.2.3 Neutron Radiography

Rods will be neutron-radiographed over the full length. Based on prior experience, these radiographs will reveal the axial location and the physical state of the pencils and the absorber pellet columns and in particular will reveal the significant internal component integrity and if any accumulation of debris has occurred.

3.10.2.4 Full-length Axial Spectral Gamma Scanning

Rods will be gamma scanned to qualitatively assess rod-to-rod variation in cladding activation and hence neutron fluence, and to characterize the axial distribution of the activation. Spectral analysis will be performed to separate component contributions which will correlate with radiography indications of pencil interface locations.

3.10.2.5 Rod Puncture, Gas Recovery, and Gas Sample Analysis and Void Volume Measurement

Rods will be punctured, and the plenum gas expanded into a known volume so as to measure its quantity. The gas will be analyzed for constituents of helium, the isotopic ratio (^{3}He / ^{4}He), and tritium activity.

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Finally, the rod will be back-filled with known quantities of noble gas to assess the void volume. This step is necessary so that the operating internal pressure can be estimated and compared to limits and design calculations.

3.10.3 DESTRUCTIVE EXAMINATIONS AND TESTS

The destructive examinations will vary from rod to rod, but overall will include the following:

- Characterization of the axial distribution and level of lithium burnup in the pellets.
- Characterization of the axial distribution of tritium and protium concentrations in the components; and characterization of the helium concentration in the pellets.

- Metallographic and microscopic evaluation of the getters and of the cladding, including the barrier layer.
- Dimensional and physical characterization of major component material: cladding, getter, pellets and liners.
- Measurement of the gettering rate on selected getter samples, using the rate test established for as-fabricated getters for comparison of the as built and post irradiation gettering rate.
- Measurement of the tritium permeation rate on selected cladding samples, using the tritium permeation rate test established for as-fabricated cladding, including archive sample measurements.

3.10.4 TPBAR FUNCTIONAL REQUIREMENTS AND PLANNED PIE

The PIE results will characterize the performance of TPBARs exposed to a full pressurized water reactor operating cycle. The information can be used to confirm adequacy and margins of the TPC TPBAR configuration via a comparison of the TPC and LTA TPBAR designs. The applicability of the planned examinations is further described in the next subsections.

3.10.4.1 Tritium Production (< 1.2 gram per TPBAR)

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A second, less accurate, estimate of the tritium production can be gained by characterizing the axial distribution of the lithium burnup and then integrating that distribution along the rod length. The burnup is estimated by measuring lithium isotopic ($^6Li/{}^7Li$) ratios in the irradiated pellets and comparing to the same ratios for the unirradiated archive material. Axial distribution of lithium burnup in a rod can be correlated with companion axial distributions of tritium and protium and helium concentrations in the components. These correlations augment analytical modeling of tritium production, release, gettering, and permeation, and protium ingress from the coolant.

3.10.4.2 Tritium Permeation Release < 1 curie per TPBAR per year

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3.10.4.3 Physical and Dimensional Stability of the Lithium Aluminate Absorber Column

Absorber column pellet and pencil integrity will be verified from the full-length neutron radiography, which will detect gaps in the column or accumulations of pellet debris. In addition, pellet physical condition will be assessed from ceramography and from observations made during section disassembly for component analyses.

Another functional requirement is that a maximum pellet GVR in the lithium aluminate pellet is not exceeded. By measuring the distribution of lithium burnup in the LTA pellet, the achieved GVR can be inferred. The axial gamma scanning will provide qualitative assurance that the axial maximum GVR can be expected to apply to all the rods.

3.10.4.4 Cladding Structural Integrity

Cladding integrity and thus retention of the tritium/helium will be assessed primarily from the results of plenum gas sampling. In addition, optical metallography and microscopy will be used to assess the metallurgical condition of the cladding. These same examinations will also determine the degree of internal and external cladding corrosion.

3.11 TPBAR SURVEILLANCE

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Because the operation of TPBARs in a full-scale tritium production core is the first application of a large number of TPBARs, a surveillance program is proposed to confirm satisfactory operation of the host plant with TPBARs and to monitor the integrity of TPBARs. The surveillance will be implemented during the first and second cycles of operation with production quantities of TPBARs and will include a surveillance inspection of a representative number of rods discharged after the first two cycles. Surveillance is intended to enable an acceptable inference regarding the satisfactory performance of TPBARs in a fuel cycle as required by the Standard Review Plan (SRP) for new fuel designs.

The surveillance program will include the following:

- During plant operation, periodic review of the reactor coolant activity measurements taken as part of the plant operation. Specifically review the tritium activity data for tritium concentration in the reactor coolant system measured during normal monitoring of the RCS chemistry to assure that the cumulative rod-average tritium release is less than 1 Ci/TPBAR/year.
- During plant operation, periodic reviews of the routine critical boron and in-core instrumentation measurements to compare the reactivity and power distribution of the production core with predictions.
- Post-irradiation examination of the TPBARs consisting of:
  - Visual examination (in the spent fuel pool) of approximately 5 to 10% of the TPBAR cluster assemblies for any evidence of wear or corrosion, loss of structural integrity, or other anomalies. The examinations will be performed after each of the initial two production cycles.

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  - After the first production cycle, a number of irradiated TPBARs shall be shipped to a DOE-specified site for additional post-irradiation examinations.

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Unusual occurrences during tritium production core operations attributable to the TPBARs may lead to additional TPBAR-related surveillance.

Surveillance of the in-reactor performance will be performed by the utility licensee for the TPC fuel cycle. The post-irradiation surveillance will be performed by DOE.

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3.12 SUMMARY AND CONCLUSION

The TPBAR as evaluated meets accepted and conservative criteria as a core component in a Westinghouse 17 X 17 type fuel assembly inserted in a PWR. The primary functions of TPBARs located in guide thimble tubes which are not under a CRDM are:

- To absorb neutrons as part of the fuel cycle reactivity control, (Section 2.4) and
- To produce and contain tritium. (Section 3.5)

The TPBARs perform their function with acceptable margin-to-failure during normal operation and in conjunction with design-basis accidents:

- As a core component, the TPBAR does not initiate or increase the severity of an accident but has the potential to affect the radiological consequences of some accidents.
- The consequences of TPBAR cladding failure have been evaluated and can be accommodated by other systems. (Section 3.5)
- The TPBARs are compatible with Westinghouse 17 X 17 assemblies operating in a high power density (up rated) core of a large 4 loop Westinghouse PWR. They are attached to standard fuel assembly base plates, are inserted in guide thimbles, and are compatible with the fuel assemblies. (Section 2.4)
- Analysis and comparison with equivalent core component assemblies has shown that the TPBAR will not fail during normal operation and Condition I through IV events with the exception of a Large Break LOCA. During the Large Break LOCA, failures of TPBARs are assumed to occur under EOL internal pressure. (Section 3.4)
- The tritium release from TPBARs can be accommodated by the plant systems. The enveloping tritium releases provided as input to the tritium release consequence evaluations are considered conservative. (Section 3.7)

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- TPBARs use materials with known and predictable characteristics in reactor performance and are compatible with the reactor coolant system (RCS). (Section 3.8)
- All significant consequences of assumed TPBAR failures (without identifying failure mechanism) were considered during normal operation and accident conditions and found to be acceptable. (Section 3.4)
- The thermal-hydraulic evaluation has shown that TPBARs operate within established thermal-hydraulic criteria. (Section 3.6)

The evaluation of the TPC TPBARs incorporates the methodology developed for the LTAs irradiated in the Watts Bar Nuclear Plant and extends this methodology to a Tritium Production Core while at the same time incorporating comments raised from the review of the LTA design.

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SECTION 4 CONCLUSIONS/NO SIGNIFICANT HAZARDS EVALUATION

4.1 CONCLUSIONS

Evaluations and analyses were performed to:

- Design a TPC which incorporates a full complement of TPBARs; that is, a design which employs a cluster of TPBARs in every possible core location (every fuel assembly except control rod locations). The goals of this design are to maximize tritium production and maintain, to the extent possible, the same key accident analysis input parameters as currently exist for the reference plant.

- Evaluate the impact of the TPC on a conservatively representative PWR (the "reference plant") design, using the NRC's Standard Review Plan (NUREG-0800) as a guide for that evaluation.

- Evaluate the impact of the reference plant parameters on the TPBAR design.

In addition to defining the impact that the TPC has on the reference PWR (and vice versa), this topical report provides a roadmap to define the analyses and tests needed to support a license amendment request.

The results of the evaluations and analyses performed establish the feasibility of implementing a TPC at a typical PWR. No safety issues were identified which would preclude the implementation. Deviations from the current reference plant key accident analysis input parameters were few, and had no overall effect on the results of the safety analyses performed for the topical report. The only potential modification to the reactor plant hardware identified for the tritium production reactor is the spent fuel cooling system modification, which may be necessary to accommodate the additional heat load. No other hardware modifications were identified to support the TPC. Post-irradiation handling, transportation, and processing are not addressed in this report, since this study concludes once the fuel assemblies containing irradiated TPBARs are transferred to the spent fuel pool.

Table 4-1 summarizes the impacts of installing a tritium production core in the reference plant.

Impact of the TPC on the Reference Plant:

Table 4-1 separates the impact into three distinct categories:

- "No further evaluation for LAR" denotes areas that have been reviewed and are clearly not affected by the TPC.

- "Confirming check recommended for LAR" defines areas that were either not affected or were affected only slightly by the TPC, for the reference plant. It should be noted that the core design for the reference plant was selected to minimize impact on the key accident analysis input parameters, NSSS operating conditions (temperatures, pressures, and flows)

and NSSS design transients. If, for some reason (e.g., fuel usage optimization), this strategy were to be modified for the implementing plant, such that key analysis input parameters changed significantly, analyses in these areas could be necessary.

- "Plant-specific evaluations required" is a self-explanatory phrase used in the third category of effort. In this category, the impact of the TPC was significant, or has the potential to be significant, depending on the plant selected, such that analyses or evaluations will be required. Table 4-1 very briefly summarizes the nature of the evaluation or analysis required. The corresponding topical report sections provide more detail on the analyses required and the factors that influence the analyses.

As a general conclusion, the areas affected most significantly by the TPC for the reference plant are:

- the core design, for which the number of feed assemblies is increased from 84 to 140 (equilibrium cycle) and the fuel enrichments are increased;

- the spent fuel pool temperatures, which increased by approximately 21°F for the reference plant as a result of the additional fresh fuel necessary for the TPC design; and

- the increase in liquid waste from approximately 10 system volumes (current reference plant discharge) to approximately 17.5 system volumes per year, and in some cases increasing to 40 volumes per year, primarily resulting from the conservatively assumed failure of two TPBARs. This is equivalent to approximately 2% of the applicable plant release limits, in terms of limiting dose to members of the public (and less than 10% of the limits, for the case of two failed TPBARs).

Impact of the Reference Plant on the TPBAR Design:

The TPBAR as evaluated meets accepted and conservative criteria as a core component in a Westinghouse 17 X 17 type fuel assembly inserted in a PWR. The primary functions of TPBARs located in guide thimble tubes which are not under a CRDM are:

- To absorb neutrons as part of the fuel cycle reactivity control. (Section 2.4)
- Produce and contain tritium. (Section 3.5)

The TPBARs perform their function with acceptable margin-to-failure during normal operation and in conjunction with design-basis accidents:

- As a core component, the TPBARs do not initiate or significantly increase the severity of any accident.

- The consequences of TPBAR cladding failure have been evaluated and can be accommodated by other systems. (Section 3.5)

- The TPBARs are compatible with Westinghouse 17 X 17 assemblies operating in a high power density (uprated) core of a large 4 loop Westinghouse PWR. They are attached to standard fuel assembly base plates, are inserted in guide thimbles, and are compatible with the fuel assemblies. (Section 2.4)

- Analysis and comparison with equivalent core component assemblies has shown that the TPBAR will not fail during normal operation and Condition III and IV events with the exception of a Large Break LOCA. During the Large Break LOCA, failures of TPBARs are assumed to occur under EOL internal pressure. (Section 3.4)
- The tritium release from TPBARs can be accommodated by the plant systems. The enveloping tritium releases provided as input to the tritium release consequence evaluations are considered conservative. (Section 3.7)

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- TPBARs use materials with known and predictable characteristics in reactor performance and are compatible with the PWR system. (Section 3.8)
- All significant consequences of assumed TPBAR failures (without identifying failure mechanism) were considered during normal operation and accident conditions and found to be acceptable. (Section 3.4)
- The thermal-hydraulic evaluation has shown that TPBARs operate within established thermal-hydraulic criteria. (Section 3.6)

The evaluation of the TPC TPBARs incorporates the methodology developed for the LTAs irradiated in the Watts Bar Nuclear Plant and extends this methodology to a Tritium Production Core while at the same time incorporating comments raised from the review of the LTA design.

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4.2 DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Introduction:

Consistent with standard Westinghouse reload analysis methodology, the existing reference plant safety analyses were reviewed to determine whether any of the key safety analysis input parameters are affected by the TPC. For a given accident, if all safety related reload parameters are bounded by those assumed in the existing analysis, then that analysis continues to define a valid licensing basis for the plant. When any safety related reload parameter is not bounded, further evaluation is necessary. Accident analysis is also performed if there are any changes to the reactor plant systems or the control or protection systems, whether as a result of the TPC or the associated reload core design.

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

Evaluation of Probability:

The presence of the TPC does not increase the probability of an accident previously evaluated. The following is a listing of the accident analysis areas. The topical report evaluations demonstrate that there is no impact of the TPC on the systems, components, or conditions that lead to an increased probability of these events occurring:

- Increase or a decrease in heat removal by the secondary system (Section 2.10).
- Decrease in reactor coolant system flowrate (Section 2.3.4.4).
- Startup of an inactive reactor coolant pump at an incorrect temperature (Section 2.3.4.4).
- Uncontrolled RCCA bank withdrawal, an RCCA misalignment, or an RCCA ejection (Sections 2.3.5 & 2.7.4).
- CVCS malfunction that results in a decrease in boron concentration in the reactor coolant (Secti 9.1).
- Inadvertent operation of the ECCS during power operation (Sections 2.6.1 & 2.7.2).
- Steam generator tube failure (Sections 2.3.4.1 & 2.5.1).
- LOCA resulting from pipe break with the reactor coolant pressure boundary (Section 2.5.1).
- Pressurizer or safety relief valve malfunction (Section 2.5.2).
- Inadvertent loading (Section 2.15.3).

Evaluation of Consequences:

The following key factors are noted with respect to the reference plant accident analyses:

1. No changes are identified in the nominal plant operating conditions (power, coolant temperature, pressure, and flow rate).

2. No changes to the reactor core thermal hydraulic characteristics or power peaking factors, which could affect the core thermal limits, are identified as a result of the TPC. Therefore, the plant thermal limit protection system setpoints do not change as a result of the TPC.

3. The nuclear design and fuel rod design calculations performed for the TPBAR reload core design have identified the following safety analysis parameters as being outside of the bounds of the current applicable reload safety analysis parameters:

   - The least-negative Doppler-only power defect at the beginning of the cycle has been reduced from 998 pcm to 920 pcm.

   - Although not due to the presence of the TPC, the maximum fuel pellet average and surface temperatures have been increased slightly (maximum of 21°F). These fuel pellet temperature increases reflect the use of a thinner IFBA coating than in the reference plant core design, which is more benign with respect to fuel rod internal pressure, but the slightly larger pellet-to-cladding gap does increase the predicted fuel temperature.

4. Fuel assembly loading error concerns also apply to burnable absorbers; therefore, misloading scenarios were examined for the TPC, specific to the burnable absorbers.

5. The tritium conservatively assumed to be released during the large break LOCA event is the entire inventory of tritium in all of the TPBARs, generated as of the end of the fuel cycle.

Due to the reduction in the Doppler power defect noted above, the two accidents which are sensitive to this parameter (uncontrolled RCCA bank withdrawal from subcritical and BOL RCCA ejection) were reanalyzed. In addition, the increase in the maximum fuel temperature prompted the reanalysis or reevaluation of the following events:

- Complete loss of forced reactor coolant flow

- Startup of an inactive reactor coolant pump at an incorrect temperature

- RCCA ejection

- Reactor coolant pump shaft seizure.

In addition, the excessive feedwater event initiated from hot zero power conditions was reanalyzed, since it relies on the results from the uncontrolled RCCA withdrawal from subcritical event.

The details and results of these non-LOCA transient analyses and evaluations are presented in Section 2.15.2 of this topical report. In all cases, the acceptance criteria are met, and the conclusions of the FSAR for the reference plant are unaffected.

A spectrum of misloading scenarios was evaluated. As concluded in topical report Section 2.15.3, a loading error is unlikely, due to administrative procedures implemented during core loading. If a loading error should occur, however, power distribution perturbations large enough to challenge fuel limits and cause significant fuel failures would be readily detectable by incore instrumentation during startup testing.

The steam generator tube rupture analysis for the reference plant was evaluated and determined to be unaffected by the TPC (Section 2.15.4), since there are no changes to the input parameters that affect the outcome of the event.

The effects of the TPC on large and small break LOCA analyses for the reference plant were evaluated. As reported in Section 2.15.5, axial and radial power distribution effects and swelling and burst effects were examined for the impact of the TPC. In all cases, the effects were determined to be insignificant, even though all the TPBARs are assumed to fail for large break LOCA. The TPC does not affect post-LOCA sump boron concentration or the time to hot leg recirculation switchover, since the core design for the TPC does not require a change to boron concentrations in ECCS components such as the RWST or the accumulators.

The impact of the TPC on the radiological consequences analyses for the reference plant, for LOCA, fuel handling accident, RCP rotor seizure, and rod ejection accidents was evaluated in detail in Section 2.15.6, since those accidents involve release of activity from damaged fuel rods. In addition, the radiological consequences of TPBAR permeation and the failure of two TPBARs during normal operation were evaluated. It was determined that there is no adverse impact on offsite thyroid doses, but there is an increase of up to 10 percent (with respect to current calculated values) in the calculated offsite whole body doses resulting from the revised core management. However, the increase in whole body doses is not significant with respect to the available margin and remains below dose acceptance limits for any of the design basis accidents. The control room whole body dose and thyroid dose also increase due to the TPBARs. In order to achieve an acceptable control room thyroid dose for the reference plant with the TPC, it would be necessary to reduce the permissible post-LOCA recirculation leakage to values that are more typical of other plants.

The effect of the TPC on the ATWS event was determined to be benign, improving the response of the reference plant to the limiting ATWS event (Section 2.15.7).

Conclusions:

The results of the accident analyses and evaluations performed demonstrate that operation with a TPC does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Normal Operation:

Under normal operation, the TPBAR functions in the reactor core are much the same as the conventional Burnable Poison Rod Assemblies (BPRAs) except that the TPBAR also functions to produce and retain tritium. To accommodate the tritium producing function, changes to the reference plant core design and plant hardware are required, and these are summarized in Section 4.1. Because of the tritium producing function, fuel enrichments need to be increased to compensate for the additional poison in the TPBARs (relative to BPRAs). A comparison of the BPRAs and TPBARs shows that both consume a poison (B^{10} and Li^{6}) and both produce He^{4} which results in an internal pressure buildup over the operating cycle.

The normal functioning of a TPBAR results in a small change to operating parameters, in that some permeation of tritium is assumed to occur without TPBAR failure. This release of tritium to the reactor coolant has been analyzed and determined to have minimal effects (see Section 2.11 of this report). The other aspect of normal TPBAR operation is how it functions as part of the core reactivity. TPBARs burn out at a slower rate than discrete burnable absorbers; this slow burnout is an advantage for reactivity control.

Besides the permeation of tritium through the cladding and into the RCS, the TPBARs differ from the BPRAs in the following areas:

a) The TPBARs contain materials not found in the BPRAs ($LiAlO_2$, nickel plating, aluminide coating, tritium, He^{3}). These materials and their interactions have been assessed (Appendix A and Section 3.8).

b) The TPBARs produce He^{3}, a strong neutron absorber, from the natural decay of tritium. During normal reactor operation, the He^{3} quickly absorbs a neutron and converts back to tritium and the He^{3} inventory remains low. During an extended mid-cycle shutdown, the He^{3} content will increase in the TPBARs. During restart of the reactor, after an extended shutdown, the He^{3} converts back to tritium. The effects of the He^{3} buildup and the consequences of a potential relocation of the He^{3} have been evaluated and found to be benign.

c) The TPBARs are removed from the fuel assembly, packaged, and shipped to a DOE facility. This activity will need to be addressed at a later time and is outside the scope of this report.

The differences between the BPRAs and the TPBARs have been assessed for normal operation, and no new accidents have been identified. TPBARs do not affect other plant initiating events or safety functions in ways different than BPRAs during normal operation.

TPBAR Failure:

Failure of TPBARs is one of the scenarios to consider in order to determine the likelihood that TPBARs will create the possibility of a new or different accident from any accident previously evaluated. The first question to be investigated is the potential for TPBAR failure. The evaluations and analyses documented in this topical report demonstrate that:

- The only plant design basis condition that can credibly result in multiple, widespread TPBAR failures is the Large Break Loss of Coolant Accident (LBLOCA).

- For all other operating conditions, it is conservative to assume that two TPBARs fail, such that there is a breach of the TPBAR cladding.

The second aspect to be determined is the impact of TPBAR failures. Does such a failure lead to the loss of a safety-related function or does it create a new type of initiating event? Does it cause other safety-related systems or components to fail? If the answers to these questions are negative, then it would seem apparent that the TPC does not create the possibility of a new or different kind of accident from any accident previously evaluated.

1. *Does such a failure lead to the loss of a safety-related function or does it create a new type of initiating event?* The safety-related function performed by the TPBARs is (in combination with the fuel and the plant control system) to provide an appropriate level of reactivity to keep the reactor in a safe state, in conjunction with the presence of the soluble boron and control rods. To perform this safety-related function, the TPBARs must absorb neutrons as predicted by design, affect the flux and power distributions in an assembly and in the core as predicted by design, and be placed in the correct core position. The only neutron absorbing component of the TPBARs is the Lithium-6 located in the ceramic lithium aluminate pellets. Testing and analyses (see Section 3 of this topical report) have demonstrated that only a small portion of the Lithium-6 in the TPBARs is lost when TPBARs fail. Therefore, the TPBARs, even for failures involving a breach of the cladding, continue to position Lithium-6 in the core to control the core reactivity. Mislocation of TPBARs in the core is addressed in 2.15.3; it is determined that the incore system of movable flux detectors, which is used to verify power shapes during startup and throughout the cycle, is capable of revealing enrichment errors or misloadings which would cause the kind of substantial power distribution perturbation that would be necessary to induce significant fuel rod failures. The failure of TPBARs does not lead to the loss of any other safety-related function.

   A Failure Modes and Effects Analysis (FMEA) was performed to evaluate the potential consequences of the failure of the TPBARs and each of the TPBAR components. The FMEA is described in Appendix A. No new or different kinds of accidents were identified that were not evaluated for previous reactor core designs. Five failure modes were identified that had the potential to result in the inability of the TPBARs to perform their safety function (i.e., any potential failures that result in the core not performing as predicted by core design to the extent there is a potential for not maintaining the core in a safe state.) These five potential failure modes are:

- misplacement of multiple fuel assemblies in the core

- multiple TPBARs not loaded

- missing multiple pencils

- lithium loading error affecting multiple TPBARs

- inadvertent operation of TPBARs for a second cycle.

These five potential failure modes are all mitigated by administrative controls used during manufacturing, refueling operations, and loading fuel into the core. Additionally, errors sufficiently large that fuel design limits are exceeded, are detectable by Technical Specification requirements for startup and flux map surveillance. Therefore, the risks associated with these potential failure modes are considered acceptable.

2. *Does the failure of TPBARs cause any safety-related components or systems to fail?* For Large Break LOCA, all TPBARs are conservatively assumed to fail. The results of this event are addressed in detail in Sections 2.15.5 and 2.15.6 of this topical report and are not significantly more severe due to the TPBARs. It is anticipated that, for a typical PWR, acceptable consequences will be demonstrated for Large Break LOCA and for all other safety analyses. The presence of the TPBARs in place of conventional burnable absorbers does not alter the nature of the LOCA event or any other accident. In fact, the failure of a TPBAR is radiologically bounded by the failure of a fuel rod, and is less likely, due to the protected location of the TPBARs within guide thimbles. That protected location practically precludes the interference of TPBARs with adjacent fuel rods, and mitigates the interference of TPBARs with post-accident core cooling (see Section 2.15.5).

Conclusions:

As part of a License Amendment Request, it is recommended that existing plant-specific procedures be reviewed to determine if there is any impact on those procedures of operation with the TPC. The analyses and evaluations performed demonstrate that, for the reference plant, existing plant systems are capable of supporting operation of the plant with the TPC. Although review of plant procedures is part of the LAR, there was nothing in the results of the analyses and evaluations performed to indicate that new, special procedures or tools are required for pre-irradiation handling, plant operation, and installation of the TPBARs in the spent fuel pool.

This report documents that the possibility of an accident which is different from any already in the reference plant FSAR is not created. The core design meets applicable design and performance standards, and ensures that pertinent licensing basis acceptance criteria are met. These standards and criteria are referenced throughout the body of this report. The demonstrated adherence to these standards and criteria precludes new challenges to components and systems that could introduce a new type of accident. Specifically, the mechanical changes will not create the possibility of an accident of a different type than any previously evaluated in the FSAR. All design and performance criteria will continue to be met

and no new single failure mechanisms have been created, as documented in this report, nor will they cause the reactor to operate in excess of pertinent design basis operating limits. Therefore, the possibility of an accident of a different type than any previously evaluated in the FSAR has not been created.

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

As noted above in the response to the previous two questions, it is anticipated that, for TPC implementation, the effects of the TPC on the safety analyses are mostly insignificant. In the one case for which the TPC has a significant adverse effect on the safety analysis results (large break LOCA control room dose), margin to the acceptance limit still exists for plants with typical permissible post-LOCA recirculation values, without any modifications to the plant's Technical Specifications. Based on the analyses provided in Section 2.15, there is not a sifnificant reduction in the margin of safety.

As noted in Section 2.16 of this topical report, there are only two anticipated changes to the plant's Technical Specifications resulting from implementation of a TPC. The first is the Spent Fuel Assembly Storage (TS 3.7.17 in NUREG-1431) and its accompanying "Design Features" description. The second potential change is to the Cold Overpressure Mitigation setpoints and associated Appendix G changes. For recent-vintage Technical Specifications, these setpoints appear in the Pressure/Temperature Limits Report, rather than in the plant's Technical Specifications.

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Table 4-1 TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
Ch. 2: Site Characteristics	No Impact	No further evaluation for LAR.	2.2
Ch. 3: Design of Structures, Components, Equipment & Systems	No impact, except: 3.9.1, 3.9.2, 3.9.3, 3.9.4, 3.9.5, 3.11	No further evaluations for LAR, except as listed in subsections below.	2.3
3.9.1 Special Topics for Mechanical Components	No impact for the tritium production reactor, due to unchanged NSSS performance parameters (RCS & FW pressure, temp., flow) and unchanged reactivity characteristics	Confirming check recommended for LAR.	2.3.2
3.9.2 Dynamic Testing & Analysis Of Systems, Components & Equipment	Negligible impact, due to slight decrease in best estimate RCS flow and corresponding decrease in RV inlet temps.	Confirming check recommended for LAR.	2.3.3
3.9.3 ASME Code Class 1, 2 & 3 Components, Component Supports, and Core Support Structures	No impact to component structural analyses, due to unchanged RCS parameters, design transients, LOCA forces, flow-induced vibration and gamma heating rates.	Confirming check recommended for LAR, for structural analysis of components. Auxiliary components for spent fuel pit should be reviewed to confirm that design temperatures bound maximum expected temperature.	2.3.4
3.9.4 Control Rod Drive Mechanism Design	No impact, due to unchanged RCS parameters, design transients, and LOCA forces.	Confirming check recommended for LAR.	2.3.5

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
3.9.5 Reactor Internals Design	TPBARs have a minor impact on Reactor Internals T/H analysis, comparable to installing thimble plugs. No impact to component structural analyses, due to unchanged RCS parameters, design transients, LOCA forces, flow-induced vibration and gamma heating roles.	Plant specific evaluation recommended for LAR.	2.3.6
3.11 Equipment Qualification	Dose rates and integrated doses considered in WCAP-8587 bound TPC values.	Confirming check recommended for LAR.	2.3.7
Ch. 4: Reactor	No impact on 4.5.1 or 4.5.2. For remaining impacts, see subsections below.	Detailed analyses or evaluations required for LAR.	2.4
4.2 Fuel System Design	For the tritium production reactor, grid load margin compensates for increased weight due to TPBARs. No impact to fuel assembly structural integrity or LOCA/seismic loads. All fuel rod design criteria met; TPBARs had insignificant impact on meeting the criteria.	Plant-specific evaluation required for LAR, both for assessment of grid load margin, and interaction between the TPBARs and fuel assembly for LOCA/seismic analysis. Confirming check recommended for LAR to verify acceptable results.	2.4.2

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
4.3 Nuclear Design	Viable TPC designs were developed for the reference plant which achieve typical cycle energy goals, generate large amounts of tritium, and meet typical design and safety limits. The number of feed assemblies increased from ~90 to ~140.	The nuclear design analysis must be repeated in its entirety for the LAR, since it is plant-specific and dependent on utility and tritium production goals.	2.4.3
4.4 Thermal and Hydraulic Design	TPBARs have little impact on T/H design; structural integrity of assembly preserved; bypass flow remains within limits; DNB criteria continue to be met.	Confirming check recommended for LAR to verify acceptable results.	2.4.4
4.6 RCCA Rod Drop Time Evaluation	Calculated rod drop time with TPC was within the Tech Spec limit of 2.7 seconds. TPBARs had an insignificant effect on rod drop time.	Confirming check recommended for LAR to verify acceptable results.	2.4.5
Ch. 5: Reactor Coolant System and Connected Systems	SRP 5.2 not impacted, except 5.2.2. SRP 5.3 details below. SRP 5.4 not impacted, except 5.4.7.	No further evaluations for LAR, except as listed in subsections below.	2.5
5.2.2 Overpressure Protection	Cold Overpressure Mitigation (COMS) setpoint affected slightly, due to slight fluence change, leading to 10CFR50 App. G limit change.	Plant-specific evaluation of App. G limit (and potential impact on COMS) recommended for LAR.	2.5.2

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations				
SRP Chapters & Sections		**Impact Summary**	**Recommendations**	**Topical Report Section**
5.3	Reactor Vessel Materials, Pressure-Temperature Limits, and Reactor Vessel Integrity	No change to current surveillance capsule withdrawal schedule. Minor change to heatup and cooldown curves (App. G). Reference plant remains within Cat. I of ERG P-T limits. No significant impact on RT_{PTS} values through 32 EFPY.	Plant-specific evaluation of vessel fluence effects recommended for LAR. Plants close to PTS screening criteria, or with a small operating window during heatup and cooldown, or with high Cu/Ni content in vessel beltline materials, may not meet acceptance criteria.	2.5.3
5.4.7	Residual Heat Removal System	Core cooldown heat load is reduced with TPC fuel cycles. Some offsetting effects from higher head load in spent fuel pool.	Plant-specific evaluation of the net effect of TPC on RHR System cooling capability is recommended.	2.5.4
Ch. 6:	**Engineered Safety Features**	No impact on Sections: 6.1.1, 6.2.4, 6.2.6, 6.2.7, 6.5.1, 6.5.2, 6.5.4, 6.6.	No further evaluation recommended, except as listed below for impacted sections.	2.6
6.1.2	Protective Coating Systems	No impact, because no impact on post-accident EQ conditions (Topical Report section 2.3.7).	No plant-specific evaluation recommended for LAR, IF no impact on post-accident EQ conditions for candidate plant.	2.6.1
6.2.1	Containment Functional Design	No impact, because no impact on LOCA or SLB M/E releases.	Plant-specific confirmation that core stored energy (and, therefore, M/E releases) do not increase is recommended for LAR.	2.6.1, 2.6.2, 2.6.3, & 2.6.4
6.2.2	Containment Heat Removal Systems	No impact because LOCA and SLB M/E releases do not increase.	Plant-specific confirmation of no change to M/E releases is recommended for LAR.	2.6.1, 2.6.2, & 2.6.3

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations				
SRP Chapters & Sections		**Impact Summary**	**Recommendations**	**Topical Report Section**
6.2.3	Secondary Containment Functional Design	Not applicable for the tritium production reactor, because it has no secondary containment.	For plants with secondary containment, a plant-specific confirmation of no change to M/E releases is recommended for LAR.	2.6.1
6.2.5	Combustible Gas Control in Containment	Assuming total hydrogen/tritium inventory released to containment as gas, this still represents only 8.3% immediate increase in hydrogen gas from LOCA. Contribution of TPC tritium inventory to Regulatory Guide limit is less than 1% of total from all sources.	Confirming check recommended for LAR	2.6.5
6.3	Emergency Core Cooling System	No impact, because no impact on post-accident EQ conditions (Topical Report Section 2.3.7).	Confirm no impact on post-accident EQ conditions for candidate plant.	2.6.1
6.4	Control Room Habitability Systems	Increases in control room whole-body dose and thyroid dose, resulting in dose acceptance criteria being exceeded for the tritium production plant. To achieve acceptable results permissible post-LOCA recirculation leakage would have to be reduced from current value.	A plant-specific evaluation is recommended for LAR. It is anticipated that acceptable values can be achieved with typical recirculation leakage limits.	2.6.1 & 2.15.6
6.5.3	Fission Product Control Systems and Structures	Effect of TPC is to increase calculated offsite whole body doses by 6 - 10 %, due to revised fuel management. Dose acceptance limits for design basis accidents not challenged.	A plant-specific evaluation is recommended for the LAR.	2.6.1 & 2.15.6

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
Ch. 7: Instrumentation and Controls	No impact on SRPs 7.1 & 7.6. Evaluations performed (minor impact) for SRPs 7.2-7.5 & 7.7.	No further evaluation recommended, except as listed below for impacted sections.	2.7
7.2 Reactor Trip System 7.3 Engineered Safety Features System	For the reference plant core design, the tritium production core had no impact on the RTS/ESFAS setpoints, since the strategy for the Topical Report was to preserve the parameters important to safety analyses, rather than to optimize on fuel usage.	For LAR, a plant-specific core design will be prepared. If one of the goals is to optimize on fuel usage, safety analysis input parameters could change, requiring a change to the protection system setpoints. Therefore, a review of this area is recommended.	2.7.2
7.4 Safe Shutdown Systems 7.5 Information Systems Important to Safety	TPC does not affect these systems for the tritium production plant.	For the LAR, if the candidate plant employs bottom-mounted thermocouples, it is recommended that the process measurement effects for post accident monitoring be revalidated with TPBARs accounted for. If not, then no plant-specific evaluation is recommended.	2.7.3
7.7 Operational Transients/Margin to Trip	No control system setpoint revisions are required for the the tritium production plant.	For LAR, a plant-specific evaluation is recommended if: the NSSS performance parameters change, the protection system setpoints change, or the fuel reactivity changes are significant with the TPC.	2.7.4

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
Ch. 8: Electric Power	No impact, because no impact on post-accident EQ conditions (Topical Report section 2.3.7).	Confirm no impact on post-accident EQ conditions for candidate plant.	2.8
Ch. 9: Auxiliary Systems	No impact, for SRPs: 9.2.4, 9.2.6, 9.3.1, & 9.3.3.	Plant-specific evaluations, as discussed below, are recommended for LAR; no evaluations recommended for the sections listed to the left of this column.	2.9
9.1.1 New Fuel Storage 9.1.2 Spent Fuel Storage	For the tritium production plant, sufficient weight margin is available to permit the extra weight of the TPBARs, so the existing seismic analysis remains valid.	A plant-specific evaluation, both for the weight margin available, and for criticality considerations, is recommended for the LAR.	2.9.2
9.1.3 Spent Fuel Pool Cooling and Cleanup System	For the tritium production plant, there is a significant increase in spent fuel pool temperature, due to the additional fresh fuel incorporated into the TPC. Additional ALARA aspects addressed in Topical Report Section 2.12.3. Additional heat exchanger capacity for the spent fuel pool may be required.	A plant-specific detailed analysis is recommended for the LAR, since spent fuel pool capacity and the core design will vary significantly from plant to plant.	2.9.3 & 2.12.3
9.1.4 Light Load Handling System	No impact.	A plant-specific evaluation of the TPBAR weight increase effect (with respect to the normal burnable poison rod assembly) on the light load handling equipment is recommended for the LAR.	2.9.1

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
9.1.5 Overhead Heavy Load Handling System	The tritium production plant evaluation did not include spent fuel cask bridge crane, since report scope does not include removal of TPBARs from spent fuel pool. No impact on remaining system features.	A plant-specific evaluation of the impact of the TPBAR additional weight on the spent fuel cask bridge crane is recommended for the LAR.	2.9.1
9.2.1 Station Service Water System	Scoping evaluation only performed for the tritium production plant; identified a potential impact, due to increase in spent fuel pool heat load during cooldown operations.	A plant-specific evaluation is recommended for the LAR, to determine the impact of increased spent fuel pool heat load on the station service water system.	2.9.1, 2.9.3, & 2.9.4
9.2.2 Reactor Auxiliary Cooling Water Systems	For the tritium production plant, no impact on CCW heat load during refueling, because increase in heat load due to additional fuel assemblies in the spent fuel pool is offset by fewer assemblies remaining in reactor (therefore, a lower RCS heat load). During cooldown, there is an impact on the CCWS, due to increased spent fuel pool heat load.	A plant-specific evaluation is recommended for the LAR, to determine the heat load increase during plant cooldown.	2.9.4
9.2.3 Demineralized Water Makeup System	For the tritium production plant, it is expected that the increased discharge volume due to the increase in tritium levels will result in increased makeup water requirements. System will be operated more frequently, but equipment sizing and flow rates are unaffected.	A plant-specific evaluation is recommended for the LAR, since the demineralized water system design varies from plant to plant.	2.9.5

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
9.2.5 Ultimate Heat Sink	Scoping evaluation only performed for the tritium production plant, which directs a plant-specific evaluation of heat load impact on ultimate heat sink.	A plant-specific evaluation is recommended for the LAR, due to the increase heat loads from the spent fuel pool during cooldown. This increase may affect the heat removal requirement for the ultimate heat sink.	2.9.1, 2.9.3, & 2.9.4
9.3.2 Process and Post-Accident Sampling Systems	No change is needed to the sampling system, but increased sampling frequency is recommended, including tritium sampling.	Confirming check recommended for LAR to establish that no additional sample points are necessary, and to direct review of existing sample frequencies and procedures.	2.9.6
9.3.4 Chemical and Volume Control System	No impact, because NSSS parameters, SI flow requirements, and boration requirements did not change for Topical Report.	A plant-specific evaluation is recommended for the LAR, to confirm that the critical parameters do not change.	2.9.1
Ch. 10: Steam and Power Conversion System	No impact, because the NSSS parameters did not change for the Topical Report.	No plant-specific evaluation is recommended for the LAR, unless the NSSS performance parameters are modified to accommodate the TPC.	2.10

Table 4-1 (continued)
TPC Impact On Reference Plant/LAR Recommendations

SRP Chapters & Sections	Impact Summary	Recommendations	Topical Report Section
Ch. 11: Radioactive Waste Management			2.11
11.1 Source Terms	For the tritium production plant, design basis tritium sources will increase the amount of tritium discharge by a factor of five. Discharges other than tritium are not significantly affected by the TPC. The TPBARs do not affect the ability of the plant to meet regulatory requirements associated with ALARA and control of radioactive releases to the environment.	A plant-specific evaluation of the impact of the TPC is recommended for the LAR.	2.11.2
11.2 Liquid Waste Management Systems	With design basis increase in plant tritium inventory, liquid effluent tritium discharge can still be maintained well below Tech Spec limit with little change in current operating practice, except increase in number of batch releases. Additional moderating steps can be taken to reduce annual discharge. With failure of two TPBARs considered conservatively, plant operation within regulatory limits is viable, but increases in intentional activity releases are needed to maintain reasonable in-plant activity concentrations.	A plant-specific evaluation is recommended for the LAR.	2.11.3
11.3 Gaseous Waste Management Systems	Impact of the TPC is small, compared to the normal operating load and the design basis for the gaseous waste system.	Confirming check recommended for LAR.	2.11.4

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
11.4 Solid Waste Management Systems	The additional number of resin bed changes is approximately one per year (estimated solid waste activity increase is from 2000 Ci to 2600 Ci, or 30 ft^3). Consideration of two failed TPBARs results in a total of approximately 4 resin bed changes per year.	A plant-specific evaluation is recommended for the LAR.	2.11.5
11.5 Process and Effluent Radiological Monitoring and Sampling Systems	Sample and analysis frequencies will increase (due to increased tritium release), but systems and equipment are adequate.	Confirming check recommended for LAR.	2.11.6
Ch. 12: Radiation Protection	No impact on SRP 12.1. See subsections below for impact on remainder of Chapter 12.	No further evaluations for LAR, except as listed in subsections below.	2.12
12.2 Radiation Sources	Negligible impact on design basis and realistic fission and corrosion product sources. No impact on fission product source terms used for shield design, EQ, system design and accident doses. Increases in discharge of tritium, but waste management can be conducted such that regulatory limits are not exceeded.	A plant-specific evaluation of the TPC is recommended for the LAR.	2.12.2

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
12.3 & Radiation Protection Design Features 12.4 Dose Assessment	Negligible increase in annual worker radiological exposure, since recommended procedure is to adjust discharge of tritiated coolant to maintain acceptable in-plant worker exposures. Also, slight increase in worker exposure due to increased fuel and TPBAR handling (0.05% of ORE).	Confirming check recommended for LAR.	2.12.3
12.5 Operational Radiation Protection Program	Increase in tritium does not change current Health Physics Program.	No further evaluation for LAR.	2.12.4
Ch. 13: Conduct of Operations	No impact, except: 13.2, 13.3, 13.5, & 13.6.	No further evaluation for LAR, except as defined in subsections below.	2.13
13.2 Training	Utility scope; not specifically addressed for the tritium production plant in Topical Report.	For LAR, it is recommended that the plant training program be reviewed to determine if additional training is required for handling/monitoring TPBARs.	2.13.1
13.3 Emergency Planning	Utility scope; not specifically addressed for the tritium production plant in Topical Report.	For LAR, it is recommended that the emergency plan be reviewed and adjusted as appropriate.	2.13.1
13.5 Administrative, Operating, and Maintenance Procedures	Utility scope; not specifically addressed for the tritium production plant in Topical Report.	For LAR, it is recommended that the plant procedures be reviewed to assure continued adequacy for TPC conditions.	2.13.1

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
13.6 Physical Security	Specific actions are required for the host utility to protect the TPBARs prior to irradiation. Security facility approval will be coordinated between DOE/NRC.	For LAR, the utility will work with DOE and NRC on demonstrating how security requirements will be met.	2.13.2
Ch. 14: Initial Test Program	Normal startup tests are sufficient for a tritium production reactor. Core loading procedures need to be reviewed.	A plant-specific evaluation of the startup test list and potentially affected procedures (as identified in the Topical Report) is recommended for the LAR.	2.14
Ch. 15: Accident Analysis	TPC impact on the tritium production plant was minimized, by designing core with same peaking factors and boration requirements.	Plant-specific evaluations and confirmations recommended for LAR, as listed in the subsections below. The extent of the effort will depend in large part on the goals of the core design.	2.15
15.1.1 - 15.1.4 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	Key safety analysis input parameters unaffected for TPC, for most events. Change to least-negative, BOL, Doppler-only power defect necessitated reanalysis of excessive feedwater event. Acceptance criteria continue to be met and FSAR conclusions remain valid for reanalyzed case with TPC.	Confirming check recommended for LAR. If any key input parameters change (as was the case for the reference plant), reanalysis of affected events is recommended.	2.15.1 & 2.15.2.5
15.1.5 Steam System Piping Failures Inside and Outside of Containment	No impacts, because no key safety analysis input parameters changed.	Confirming check recommended for LAR.	2.15.2.5

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
15.1.5, App. A Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	No impact, since increase in tritium released to the reactor coolant is countered by increased discharge to prevent tritium buildup.	Confirming check recommended for LAR.	2.15.6.4
15.2.1-15.2.5 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve, and Steam Pressure Regulator Failure (Closed)	Insignificant impact of update to analysis technique, not due to TPC, results in slightly increased maximum fuel temperatures. Existing sensitivities demonstrate insignificant impact.	Confirming check recomm[illegible]or LAR.	2.15.2.6
15.2.6 Loss of Nonemergency AC Power to the Station Auxiliaries	Insignificant impact of update to analysis technique, not due to TPC, results in slightly increased maximum fuel temperatures. Existing sensitivities demonstrate insignificant impact.	Confirming check recommended for LAR.	2.15.2.6
15.2.7 Loss of Normal Feedwater Flow	Insignificant impact of update to analysis technique, not due to TPC, results in slightly increased maximum fuel temperatures. Existing sensitivities demonstrate insignificant impact.	Confirming check recommended for LAR.	2.15.2.6
15.2.8 Feedwater System Pipe Breaks Inside and Outside of Containment	Insignificant impact of update to analysis technique, not due to TPC, results in slightly increased maximum fuel temperatures. Existing sensitivities demonstrate insignificant impact.	Confirming check recommended for LAR.	2.15.2.6

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
15.3.1 - 15.3.2 Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	Complete Loss of Reactor Coolant Flow is the bounding of 2 events considered, and was reanalyzed for slightly increased initial fuel temperatures (not directly related to TPC). All acceptance criteria are met, and the FSAR conclusions remain valid.	Confirming check recommended for LAR.	2.15.2.7
15.3.3 - 15.3.4 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	Insignificant impact of update to analysis technique, not due to TPC, results in slightly increased maximum fuel temperatures. Shaft seizure event (which bounds shaft break) was reanalyzed, and FSAR conclusions remain valid. Insignificant dose increase, due to TPC-related change to core source terms.	Confirming check recommended for LAR.	2.15.2.7.3, 1.15.2.7.4, & 2.15.6.4
15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	Analysis is affected slightly by change to Doppler-only power defect at BOL and by slightly increased maximum fuel temperatures. Reanalysis performed - all acceptance criteria are met and the FSAR conclusions remain valid.	Confirming check recommended for LAR.	2.15.2.8.1
15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power	No impact.	Confirming check recommended for LAR.	2.15.2.8
15.4.3 Control Rod Misoperation	No impact.	Confirming check recommended for LAR.	2.15.2.8

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
15.4.4 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature	Slight effect on analysis of fuel temperature increases, but all acceptance criteria met and FSAR conclusions remain valid.	Confirming check recommended for LAR.	2.15.2.8.2
15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	No impact.	Confirming check recommended for LAR.	2.15.2.8
15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	Several misloading scenarios evaluated for reference plant. Power distribution perturbations large enough to challenge fuel limits are readily detectable by incore instrumentation during startup testing.	Core-specific evaluation recommended for LAR.	2.15.3
15.4.8 Spectrum of Rod Ejection Accidents	Analysis is affected slightly by change to Doppler-only power defect at BOL and by slightly increased maximum fuel temperatures. Reanalysis performed - all acceptance criteria are met and the FSAR conclusions remain valid.	Confirming check recommended for LAR.	2.15.2.8.3
15.4.8, App. A Radiological Consequences of a Control Rod Ejection Accident	Insignificant dose increase, due to TPC-related change to core source terms.	Confirming check recommended for LAR.	2.15.6.4
15.5.1 - 15.5.2 Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	No impact.	Confirming check recommended for LAR.	2.15.2.9

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
15.6.1 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve	No impact.	Confirming check recommended for LAR.	2.15.2.10
15.6.2 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	No impact, since increase in tritium released to the reactor coolant is countered by increased discharge to prevent tritium buildup.	Confirming check recommended for LAR.	2.15.6.4
15.6.3 Radiological Consequences of Steam Generator Tube Failure	No impact, since increase in tritium released to the reactor coolant is countered by increased discharge to prevent tritium buildup.	Confirming check recommended for LAR.	2.15.6.4
15.6.5 & App. A&B LOCAs resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	TPBAR temperatures during LOCA calculated (see Topical Report Section 3 for discussion of impact on TPBAR integrity). Effects of TPBARs on LBLOCA and SBLOCA accident progressions are not significant. Dose impact from LBLOCA with TPBARs is mainly for control room doses (large increase to thyroid, small increase to whole body). For LBLOCA dose calculation, all TPBARs are assumed to fail.	Plant-specific analyses for LBLOCA and SBLOCA recommended for LAR. Confirming checks recommended for LOCA-related events: hot leg switchover, long-term core cooling, post-LOCA sump boron concentration. Plant-specific dose calculations recommended for LBLOCA for LAR.	2.15.5 & 2.15.6.2
15.7.3 Postulated Radioactive Releases due to Liquid-Containing Tank Failures	No impact, since increase in tritium released to the reactor coolant is countered by increased discharge to prevent tritium buildup.	Confirming check recommended for LAR.	2.15.6.4

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
15.7.4 Radiological Consequences of Fuel Handling Accidents	Insignificant impact on spent fuel pool and refueling cavity temperatures and doses due to tritium release from damaged TPBARs.	Confirming check recommended (and reference to procedure for removal of damaged TPBARs) for LAR.	2.15.6.3
15.7.5 Spent Fuel Cask Drop Accidents	This accident not considered credible for reference plant, due to plant-specific equipment and procedures that preclude occurrence.	Confirming check recommended for LAR.	2.15.1
15.8 Anticipated Transients Without Scram	TPC designs improve the response of the reference plant to the limiting ATWS event.	Confirming check recommended for LAR.	2.15.7
Ch. 16: Technical Specifications	Potential changes identified for Spent Fuel Assembly Storage for reference plant, as well as some of the bases sections. Appendix G limits and PORV setpoints require revision; however, these appear in the PTLR (not Tech Specs), for the reference plant.	Plant-specific review of tech specs required for LAR.	2.16
Ch. 17: Quality Assurance	SRP 17.3 is not applicable to the TPC Program. See impacts for SRPs 17.1 and 17.2 in the subsections below.	No further effort recommended for LAR, except as listed in subsections below.	2.17

Table 4-1 (continued) TPC Impact On Reference Plant/LAR Recommendations			
SRP Chapters & Sections	**Impact Summary**	**Recommendations**	**Topical Report Section**
17.1 Quality Assurance During the Design and Construction Phase	The Westinghouse Quality Management System (QMS) complies with SRP 7.1. Design activities for the TPC have been and will be controlled in accordance with the QMS, with TPC-specific implementing procedures created when necessary.	Reference plant results apply to LAR. TPBAR-specific procedures to be created as necessary.	2.17.2
17.2 Quality Assurance During the Operations Phase	This is utility scope and was not addressed in the Topical Report.	Plant-specific evaluation recommended for LAR.	2.17.2
Ch. 18: Human Factors Engineering	No Impact	No further evaluation for LAR.	2.18

Table 4-1 (continued)
TPC Impact On Reference Plant/LAR Recommendations

Checklist of Items to Address for LAR, not within Topical Report Scope
1. Removal of the TPBARs from the Spent Fuel Pit: a) Check capability of Overhead Heavy Load Handling System to remove TPBARs b) Evaluate dose impact of handling TPBARs during removal and alignment operations 2. Evaluate equipment and dose issues for transportation and dry cask storage of TPBARs 3. Documents to review for potential impact of TPBARs (operational, monitoring, handling): a) Pressure-Temperature Limits Report (PTLR) b) Core Operating Limits Report (COLR) c) Plant Procedures d) Operator Training e) Emergency Planning

Appendix A
TPBAR Tritium Production Core (TPC)
Failure Modes and Effects Analysis (FMEA)

All information on pages A-1 through A-12
has been removed from this document.