March 31, 1999 SECY-99-099

FOR: The Commissioners

FROM: William D. Travers /s/
Executive Director for Operations

SUBJECT: TRANSMITTAL OF THE STAFF'S SAFETY EVALUATION OF DOE'S TOPICAL REPORT ON THE TRITIUM PRODUCTION CORE

PURPOSE:

This paper transmits to the Commission the NRC staff's evaluation of the Department of Energy's (DOE's) topical report on the tritium production core (TPC).

BACKGROUND:

The staff has been providing review and consultation services to assist DOE under the provisions of the memorandum of understanding between the NRC and DOE dated May 22, 1996.

DOE has developed an alternative design for burnable absorber rods using lithium-6 in place of boron-10 to control reactivity in the core and to also produce tritium. DOE expects that approximately 3300 tritium-producing burnable absorber rods (TPBARs) would be included in each core reload of a TPC.

DOE has prepared a topical report addressing how the inclusion of a full-core load of TPBARs in a reactor core affects nuclear plant systems, safety and components analyses, and performance of a representative commercial light-water reactor (CLWR). DOE intended that its topical report

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would establish an envelope of design, methodology, and analysis, serving as a guide and referenceable document for plant-specific applications to incorporate TPBARs in any CLWR design in the United States for the production of tritium.

In SECY-96-212, “Review of Department of Energy’s Proposal for Tritium Production in Commercial Light-Water Reactors,” dated October 3, 1996, the staff described the process by which it intended to conduct its review of DOE’s proposal for producing tritium in CLWRs. Regarding the production phase of DOE’s CLWR tritium program, the staff proposed to review the topical report submitted by DOE concerning the TPC, to transmit the results of its review to the Commission before issuing it, and to place a notice in the *Federal Register* announcing the availability of the staff’s safety evaluation.

In its staff requirements memorandum (SRM) dated December 10, 1996, the Commission approved the staff’s proposed review approach and, in addition, directed that public meetings be held in the vicinity of each reactor facility that undertakes irradiation of TPBARs for the production of tritium.

DISCUSSION:

On July 30, 1998, DOE submitted its TPC topical report for the staff’s review. This topical report presented the results of DOE’s systematic and comprehensive evaluation of the impact of using TPBARs in all available core locations on all aspects of reactor and balance-of-plant design, using the staff’s review criteria in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (SRP), as guidance. The topical report addresses each section of the SRP and contains an assessment of the impact of the TPC on a representative plant safety analysis report. The staff has reviewed DOE’s TPC topical report and has prepared the attached safety evaluation documenting its review.

CONCLUSIONS:

The staff has reviewed DOE’s topical report on the TPC and the related supporting information. Many technical issues have been satisfactorily addressed in the DOE topical report, as documented in the staff’s safety evaluation. During its review, the staff identified a number of interface issues that will require changes to the plant safety analysis report and that must be reviewed by the staff before the staff can determine the acceptability of irradiating a full-core load of TPBARs in any particular reactor facility. Therefore, the staff concludes that if any licensee wishes to undertake production irradiation of TPBARs, it must first submit an application for an amendment to the individual facility operating license for authorization to conduct such irradiation. Such application must address the plant-specific interface issues identified in the staff’s safety evaluation and must include the necessary changes to the plant technical specifications located in Appendix A to the operating license.

RECOMMENDATIONS:

1. That the Commission note that the staff intends to approve the attached Safety Evaluation 10 days after the date of the paper unless otherwise directed by the Commission. Action will not be taken until the SRM is received. The staff intends to issue the attached safety evaluation as NUREG-1672. At that time, the staff will prepare a *Federal Register* notice announcing the availability of the staff's safety evaluation related to DOE's topical report on the TPC. We consider this action to be within the delegated authority of the EDO.

2. That the Commission note that the staff intends to hold public meetings in the vicinity of each host facility before the irradiation of TPBARs to produce tritium.

COORDINATION:

The Office of the General Counsel has no legal objection to this paper.

The staff briefed the Advisory Committee on Reactor Safeguards during the March 1999 meeting and discussed the conclusions in its safety evaluation.

original /s/ by
Frank J. Miraglia for

William D. Travers
Executive Director
for Operations

Attachment: Safety Evaluation

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Safety Evaluation Report

related to the Department of Energy's
topical report on the tritium production core

Project No. 697

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

March 1999

ABSTRACT

The Department of Energy (DOE) is responsible for establishing the capability to produce tritium, an essential material used in U.S. nuclear weapons, by the end of 2005, in accordance with a Presidential decision directive.

Under the terms of the Joint DOE/NRC Memorandum of Understanding of May 22, 1996, NRC is providing review and consultation services to assist DOE in assessing and resolving technical and licensing issues associated with DOE's proposal for the production of tritium in a commercial light-water reactor (CLWR).

DOE has submitted a topical report on the tritium production core that systematically evaluates the impact of irradiating up to approximately 3300 tritium-producing burnable absorber rods (TPBARs) in a reactor core on all of the areas covered by the Standard Review Plan (NUREG-0800). This report is expected to be referenced by licensees participating in DOE's CLWR tritium program and to form the basis for a plant-specific application for an amendment to the facility operating license authorizing irradiation of TPBARs for the production of tritium.

The staff's review of the DOE topical report on the tritium production core and the staff's conclusions regarding the acceptability of irradiating up to approximately 3300 TPBARs in a core reload are documented in this safety evaluation. The staff has also identified a number of interface items that must be addressed by a licensee referencing the "Tritium Production Core Topical Report" in its plant-specific application for authorization to produce tritium for DOE. These are listed in Section 5 of this report.

This safety evaluation is being transmitted to the Commission before issuance.

TABLE OF CONTENTS

ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AISI	American Iron and Steel Institute
ALARA	as low as reasonably achievable
AMSAC	ATWS mitigation system actuation circuitry
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATR	Advanced Test Reactor
ATWS	anticipated transient without scram
BA	burnable absorber
BOL	beginning of life
BOP	balance of plant
BP	burnable poison
BPRA	burnable poison rod assembly
BWR	boiling-water reactor
CCWS	component cooling water system
CFR	Code of Federal Regulations
CLWR	commercial light-water reactor
CMTR	certified material test report
COL	combined license
COMS	cold overpressure mitigation system
CRD	confidential restricted data
CRDM	control rod drive mechanism
CVCS	chemical and volume control system
CW	cold-worked
DBA	design-basis accident
DBLOCA	design-basis loss-of-coolant accident
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	Department of Energy
DORT	discrete ordinate transport (code)
ECCS	emergency core cooling system
EFPD	effective full-power day
EMA	equivalent margin analysis
EFPY	effective full-power year
EOL	end of life
EPRI	Electric Power Research Institute
EQ	equipment qualification
ESF	engineering safety feature
ETR	Engineering Test Reactor
FCL	facilities (security) clearance
FMEA	failure modes and effects analysis
FSAR	final safety analysis report

GDC	general design criterion
GVR	gas volume ratio
HFIR	High Flux Isotope Reactor
HFP	hot full power
IEEE	Institute of Electrical and Electronics Engineers
IFBA	integral fuel burnable absorber
IFM	intermediate flow mixer
IGSCC	intergranular stress corrosion cracking
LB	large break
LBLOCA	large-break loss-of-coolant accident
LOCA	loss-of-coolant accident
LTA	lead test assembly
LTOP	low temperature overpressure protection
LTOPS	low temperature overpressure protection system
MCNP	Monte Carlo N-Particle (Transport Code Version 4A)
MPH	Materials Property Handbook
MOL	middle of life
MOU	memorandum of understanding
MTC	moderator temperature coeffecient
NDE	nondestructive examination
NPSH	net positive suction head
NPZ	nickel-plated Zircaloy-4
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OBE	operating-basis earthquake
ODCM	offsite dose calculation manual
OL	operating license
PIE	post-irradiation examination
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
QA	quality assurance
RAI	request for additional information
RCCA	rod cluster control assembly
RCB	reactor coolant boundary
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RHRS	residual heat removal system
RT	reference temperature
SAR	safety analysis report
SB	small break
SBLOCA	small-break loss-of-coolant accident
SBO	station blackout
SCC	stress corrosion cracking
SER	safety evaluation report
SFPCS	spent fuel pool cooling system

SGTR	steam generator tube rupture
SRM	staff requirements memorandum
SRP	Standard Review Plan
SS	stainless steel
SSE	safe-shutdown earthquake
SSWS	station service water system
TEDE	total effective dose equivalent
TGSCC	transgranular stress corrosion cracking
TMI	Three Mile Island
TPBAR	tritium-producing burnable absorber rod
TPC	tritium production core
TR	topical report
TS	Technical Specifications
TVA	Tennessee Valley Authority
UET	unfavorable exposure time
UHS	ultimate heat sink
USE	upper shelf energy
WABA	wet annular burnable assembly

1 INTRODUCTION

On July 30, 1998, the Department of Energy (DOE) submitted a report prepared by its contractor, Westinghouse Electric Company, NDP-98-153, entitled “Tritium Production Core (TPC) Topical Report,” (TPC TR), to present technical information related to production of tritium using tritium-producing burnable absorber rods (TPBARs) in a commercial light-water reactor (CLWR). As this report contained confidential restricted data, DOE submitted an unclassified version, NDP-98-181, at the same time. On December 10, 1998, DOE responded to the staff's requests for additional information dated September 29 and October 15, 1998. On February 10, 1999, DOE submitted Revision 1 to the classified and the unclassified versions of its TPC TR.

1.1 Background

Tritium, an essential material in U.S. nuclear weapons, is an isotope of hydrogen that decays at a rate of approximately 5 percent per year (a 12.3-year half-life). The United States has not produced tritium since 1988, when DOE closed its production facility at the Savannah River plant near Aiken, South Carolina. Current, short-term tritium needs are being met by recycling tritium from dismantled U.S. nuclear weapons. Resumption of tritium production will be essential for maintaining the U.S. nuclear weapons stockpile and the U.S. nuclear deterrent.

DOE's Dual-Path Strategy for the Production of Tritium

DOE is responsible for establishing the capability to produce tritium by the end of 2005, in accordance with a Presidential decision directive. DOE has selected a dual-path strategy to meet the schedule. On December 22, 1998, the Secretary of Energy announced the selection of the CLWR production of tritium as the primary path, with the accelerator production of tritium as a backup. Should DOE elect to develop its accelerator design (utilizing a tungsten target), it may pursue that option without Commission approval because the Nuclear Regulatory Commission (NRC) does not have statutory authority to regulate accelerators or DOE production facilities.

DOE proposes to produce tritium in CLWRs by contracting with the Tennessee Valley Authority (TVA) for irradiation services at TVA’s Watts Bar and Sequoyah facilities. Production of tritium in a CLWR, however, is subject to NRC statutory authority for the regulation of CLWRs.

DOE intends to complete confirmatory testing, support an NRC licensee’s request for facility operating license authorization to perform irradiation services, fabricate the first core load of targets, and develop a new extraction capability as a contingency to meet national defense requirements. Tritium would be extracted at DOE's Savannah River plant and would not involve oversight by NRC.

Joint DOE/NRC Memorandum of Understanding

On May 22, 1996, the Secretary of Energy and the Chairman of the Nuclear Regulatory Commission signed a Joint DOE/NRC Memorandum of Understanding (MOU). This MOU establishes the basis for NRC review and consultation concerning DOE's possible use of CLWRs for producing tritium. It supplements an earlier MOU between DOE and NRC (dated February 24, 1978) and relates solely to NRC's review of and consultation on DOE's proposal for tritium production in CLWRs. The MOU acknowledges that an issue exists involving the use of civilian commercial reactors to support military requirements but stipulates that NRC will not be involved, either in a policy or a technical role, in resolution of that issue. The MOU also stipulates that NRC will not be involved in the decision on whether to use an accelerator or a CLWR to produce tritium.

Under the terms of the MOU, NRC is providing review and consultation services to assist DOE in assessing and resolving technical and licensing issues associated with CLWR production of tritium (including physical security, security clearance, and environmental issues) in order to support a decision by the Secretary of Energy on the primary and backup tritium production approaches and to determine the licensing actions necessary to implement the CLWR option.

CLWR Production of Tritium

DOE has developed a design for burnable poison rods using lithium, rather than boron, in pressurized-water reactor (PWR) fuel assemblies. As a result of irradiation by neutrons in the reactor core, some of the lithium in the target rods is converted to tritium. The irradiated burnable poison rods can then be removed from the fuel assemblies and shipped to another location (Savannah River plant) for tritium extraction.

The first phase of the tritium program was a lead test assembly (LTA) demonstration, which required the approval of the NRC before implementation. The LTA irradiation was intended to serve as a confirmatory test of the design for TPBARs that DOE has developed over the past 10 years. In the safety evaluation report documenting the staff's review of DOE's LTA report, NUREG-1607, "Safety Evaluation Report related to the Department of Energy's proposal for the irradiation of lead test assemblies containing tritium-producing burnable absorber rods in commercial light-water reactors," issued in May 1997, the staff concluded that licensee(s) could not undertake irradiation of the LTAs under the provisions of 10 CFR 50.59 without NRC licensing action. Subsequently, TVA applied for an amendment to the facility operating license for Watts Bar Unit 1, authorizing irradiation of LTAs containing a total of 32 TPBARs (8 each in 4 LTAs, with 1 LTA inserted in each quadrant of the core) for one cycle. The license amendment was issued on September 15, 1997, and the reactor entered criticality for Cycle 2 on October 8, 1997. Cycle 2 ended on February 27, 1999, when the reactor shut down for its second refueling after 471.4 effective full-power days. Currently, DOE expects to ship the irradiated TPBARs to Idaho National Environmental Engineering Laboratory in July 1999, for non-destructive examination and, eventually, to Pacific Northwest National Laboratory, DOE's fabricator for the TPBARs used in the LTA demonstration, for destructive examination.

The second phase of DOE's tritium program, which is the current action and the subject of this report, involves submittal of a tritium production core (TPC) topical report to support eventual referencing by a CLWR undertaking production irradiation of TPBARs. DOE submitted its topical report on the TPC presenting technical information related to production of tritium using tritium-producing burnable absorber rods (TPBARs) in a CLWR on July 30, 1998. On December 10, 1998, DOE submitted its responses to the staff's requests for additional information dated September 29 and October 15, 1998. On February 10, 1999, DOE submitted Revision 1 to its TPC TR. The staff has prepared this safety evaluation report, documenting its review of DOE's TPC TR and is transmitting it to the Commission before issuing it.

The third and final phase of DOE's tritium program, also requiring NRC's review and approval, will be the actual production of tritium in a CLWR. DOE has entered into an agreement with TVA to produce tritium at the Watts Bar Unit 1 and Sequoyah Units 1 and 2 facilities. The NRC will review and approve any request by TVA for an amendment to the facility operating licenses for these units to authorize use of up to approximately 3300 TPBARs in each core reload for the production of tritium. The staff reviewed the TPC TR concurrently with the irradiation of the LTAs and prepared its safety evaluation on the production phase topical report before DOE's full evaluation of the LTA data. DOE has stated that, because the primary purpose of the LTA demonstration is to build confidence among prospective licensees, availability of the results of the post-irradiation examination (PIE) of the LTA demonstration is not an essential precursor to the staff's review of the DOE TPC topical report. The NRC staff agrees that it can conduct its review of the DOE TPC topical report without having the PIE results of the LTA demonstration. However, the staff may use information from the LTA demonstration in order to reach a conclusion of acceptability regarding any individual application by TVA for amendments authorizing the use of the TPC at its Watts Bar and Sequoyah facilities. A license amendment is required in order to make changes to the plant technical specifications and to address any unreviewed safety questions pertaining to such use. A request for a license amendment authorizing irradiation of burnable poison rods for production of tritium is expected to be received at the beginning of 2000. A request for a license amendment will be noticed in the *Federal Register* and will be the subject of an opportunity for a hearing. If a hearing is requested, the Commission will be notified if the staff intends to make a final finding of "no significant hazards consideration" (which would allow the amendment to become effective before the conclusion of a hearing).

The first core loading of TPBARs will be fabricated during 2002 and 2003 as part of DOE's target demonstration program, by which time the licensing activities to support CLWR production of tritium are expected to be completed. Should the CLWR production option be exercised, the TPBARs will be irradiated for one cycle, cooled, and shipped in 2005 to support the Presidential decision directive's requirement for production of the first tritium gas at the Savannah River plant by the end of 2005.

<u>SECY-96-212</u>

In SECY-96-212, the staff described DOE's proposal for the CLWR production of tritium and presented its approach for reviewing DOE's proposal under the terms of the joint MOU of May 22, 1996. The staff proposed to consider whether LTAs containing TPBARs could be irradiated under the provisions of 10 CFR 50.59 without NRC licensing action.

In its staff requirements memorandum (SRM) of December 10, 1996, the Commission approved the staff's review approach. However, the Commission directed the staff to hold a series of public meetings to give the public an opportunity to comment on the technical issues during the LTA phase and to inform the public of the staff's activities early in the evaluation process. The initial public meeting directed by the Commission was held at NRC headquarters in Rockville, Maryland on February 25, 1997, and presented the programmatic aspects of DOE's tritium program. The next public meeting directed by the Commission was held near TVA's Watts Bar facility in the summer of 1997, before the TPBAR LTAs were inserted into the reactor for irradiation. The Commission also directed the staff to hold similar local public meetings before TPBARs are inserted in any particular NRC-licensed facility for the production phase of DOE's CLWR tritium program.

1.2 Purpose

DOE has stated that the purpose of its TPC topical report is to establish an envelope of design, methodology, and analysis to be referenced by licensees participating it its program for the CLWR production of tritium.

The staff's review of the DOE TPC topical report and the staff's conclusions regarding the acceptability of irradiating up to approximately 3300 TPBARs in a core reload are documented in this safety evaluation.

The staff has also identified a number of interface items that must be addressed by a licensee referencing the DOE TPC topical report in its plant-specific application for authorization to produce tritium for DOE. These are listed in Section 5 of this report.

1.3 Scope

The staff has evaluated DOE's TPC topical report, submitted by letter dated July 30, 1998, and revised by letter dated February 10, 1999. The staff has also considered information submitted by DOE in its letters dated December 2, 1998, and January 13, 1999, responding to the staff's requests for additional information dated September 29 and October 15, 1998. Although the staff's review included the classified versions of DOE's submittals, none of the information in this safety evaluation is classified.

In selecting the reference reactor for use in preparing the DOE TPC topical report, an uprated plant with a 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 approximately 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.

The DOE TPC topical report addresses how the inclusion of a significant number of TPBARs in a reactor core affects nuclear plant systems, safety, and components analyses and performance for a representative CLWR. DOE intended that this report would serve as a guide and referenceable document for plant-specific efforts to license incorporation of TPBARs for any CLWR design the United States.

DOE did not address the activities required to remove the TPBARs from the fuel assemblies and prepare them for shipment because these activities are dependent on the fuel pool design. Therefore, the staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

Independent of its review of the DOE TPC topical report, the staff is conducting vendor-related activities with respect to quality assurance (QA) plans and fabrication inspections in order to determine compliance with the requirements of Appendix B to 10 CFR Part 50 and with 10 CFR Part 21. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

1.4 Organization of This Safety Evaluation

The format of this safety evaluation follows that of the DOE TPC topical report (NDP-98-181) as closely as possible. The staff has added Section 5 to summarize the results of its review. Section 5 of this safety evaluation also summarizes the remaining plant-specific issues that will have to be addressed in the Tennessee Valley Authority's application for an amendment to the facility operating license for Watts Bar or Sequoyah to permit irradiation of TPBARs for the production of tritium.

2 STANDARD REVIEW PLAN EVALUATION

In Section 2 of its topical report on the tritium production core (TPC), the Department of Energy (DOE) presents the results of its assessment of the effect of using tritium-producing burnable absorber rods (TPBARs) in all available core locations on all aspects of a "standard" or reference commercial light-water reactor (CLWR) design, using the Commission's review criteria ["Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (SRP) NUREG-0800] as guidance. The TPC topical report addresses each section of the SRP, and describes the input, methodology, and results relative to the TPBAR impact on the reference reactor. The majority of Section 2 of the TPC topical report is dedicated to the portions of the SRP in which there is an impact from inclusion of TPBARs; however, for the unaffected sections of the SRP, the bases for the judgement of "no impact" are provided. The TPC topical report is structured to follow the format of the SRP; i.e., 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 SRP section. DOE states that its intent was not to provide a complete safety analysis report for the tritium production core, but to provide a methodical evaluation of the effect of incorporating a full complement of TPBARs in a typical CLWR.

The staff conducted its review of the TPC topical report by evaluating DOE's assessment and responses to the staff's requests for additional information against the guidance of the SRP and other applicable regulatory guidance. During its evaluation, the staff considered whether DOE's assessment contained significant omissions of relevant SRP criteria, whether the SRP criteria were correctly interpreted, and whether other review considerations (based on operating experience occurring since the SRP sections were last revised) had been included. The NRC staff reached the conclusions discussed throughout this chapter after consideration of how the design criteria addressed in the SRP would be affected by the use of TPBARs in an operating reactor design, given the likely changes that the introduction of such a modification to the reactor core design would have on the interactions of a plant's integrated operating systems. The results of the staff's evaluation are presented below.

2.1 Introduction and General Description of Plant

In Section 2.1 of its topical report on the tritium production core, DOE addressed the topics evaluated in Chapter 1 of the SRP. Chapter 1 in a standard safety analysis report (SAR) format presents 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 Section 1.8, "Interfaces for Standard Designs." That 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 nuclear steam supply system (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 its design. Therefore, for the above stated reasons, DOE has concluded, and the staff agrees, that SRP Section 1.8 is not applicable.

Although there are no standard review plans applicable to Chapter 1 of a SAR incorporating TPBARs, there is a brief description of the reactor in SAR Section 1.2.3.1 (Reactor Core), which discusses burnable absorber rods generically as a means of suppressing reactivity at beginning of core life. DOE concludes that any future 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 a broad nature, so that the incorporation of TPBARs would not change the description. The staff agrees with DOE's assessment.

2.2 Site Characteristics

In Section 2.2 of its TPC topical report, DOE addresses the changes to the sections of Chapter 2 of a typical safety analysis report (SAR) to accommodate a full core load of TPBARs. Chapter 2 deals primarily with the physical location and characteristics of the site, not the fuel or core design. Although a SAR considers the consequences of an accidental release of liquid effluents, which are dependent on the concentration of radioactive material, (specifically tritium) assumed to be present in the spill, DOE recommends that the plant be operated with a sufficient increase in the primary coolant discharge for normal operation to ensure that the tritium concentration in liquid wastes remains within that current operation. Therefore, with this one exception, DOE concludes, and the staff agrees, that there would not be any changes to Chapter 2 of a typical SAR as a result of incorporating TPBARs.

2.3 Design of Structures, Components, Equipment, and Systems

2.3.1 Introduction

In Section 2.3 of the TPC topical report, DOE states that the design of structures, components, equipment, and systems will, for the most part, not be affected 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.

2.3.2 NSSS Design Transients (SRP Section 3.9.1) Evaluation

The NSSS design transients are used as an input for the component fatigue stress analyses of the various reactor coolant system (RCS) components [reactor vessel and internals, steam generators, RCS piping, reactor coolant pumps (RCPs), 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.

There are several parameter changes that could require modifications of the RCS design transients. DOE compared the parameters identified for the TPC with the existing parameters for the reference plant in the TPC topical report. The significant input parameters and assumptions are discussed in Section 2.3.2 of the TPC topical report.

DOE's comparison of the design operating parameters for the TPC plant versus 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

On the basis that there are no significant differences in the parameters discussed above for the TPC plant versus the reference plant, the staff concurs with the assessment in the TPC topical report that there is no need to revise the RCS design transients.

2.3.3 Dynamic Loads (SRP Section 3.9.2) Evaluation

The effect of TPBARs on hydraulic forces during a large break loss-of-coolant accident (LOCA) has been evaluated on both a best-estimate basis and on a design-parameter basis in the TPC topical report. 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 a 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 percent decrease in loop flow rate, compared to the case with no thimble plugs or TPBARs. For a typical three- or four-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 percent.

Such an increase in peak horizontal LOCA forces is considered to be small enough not to change the break size estimates, thus eliminating the need to review the American Society of Mechanical Engineers (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 these considerations, on a generic basis, DOE concludes that it is unlikely that the incorporation of TPBARs would require any reanalysis of the LOCA forces and the associated rework of the ASME Code stress analyses. However, for specific plant implementation of the TPC, a confirming evaluation based on selected RCS parameters is recommended by DOE to support a plant specific application. The results of this evaluation will provide input to the reactor vessel, reactor internals, and reactor coolant piping/supports structural analyses. The staff agrees with DOE's assessment.

2.3.4 Components (SRP Section 3.3.9) Evaluations

The effect of the TPC on the existing structural analyses of the steam generators, pressurizers, reactor coolant piping and supports, RCPs, auxiliary heat exchangers, tanks, pumps and valves, and the reactor vessel was assessed by DOE. DOE's rationale for the judgment of “no impact” and the staff’s views with respect to each of these components is discussed below:

<u>Steam Generator Components</u>

The NSSS performance parameters for the proposed TPC plant, provided in Table 1-1 of the TPC topical report, are the same as those currently licensed for the reference plant. It has been determined by DOE, and the staff agrees, that in accordance with Section 2.3.2, the plant design transients will not change as a result of 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 in the reference plant, and will continue to do so for the TPC plant.

DOE states that the conclusions of its 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. However, for a specific plant implementation, the impact concerning the RCS parameters and NSSS design transients will need to be addressed by licensees participating in DOE's program for the CLWR production of tritium to determine the effect on the steam generator structural evaluation. The staff agrees with DOE's assessment.

Pressurizer Components

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 key inputs ($T_{HOT'}$,T_{COLD} and T_{PZR}) were found to be identical for the reference plant and the TPC plant. Furthermore, the transients for the reference plant pressurizer are not affected by the TPC.

On the basis of these observations, the staff agrees with the DOE assessment that the reference plant pressurizer stress analysis envelops the TPC plant parameters. Therefore, no additional stress/fatigue/fracture mechanics analyses are required for the pressurizer components for the TPC. However, for specific plant implementation, the RCS parameters and NSSS design transients need to be addressed by licensees participating in DOE's program for the CLWR production of tritium to determine any affect on the pressurizer. The staff agrees with DOE's assessment.

Reactor Coolant System Piping and Supports

The effect of the TPC installation on the RCS piping and supports is addressed by evaluating the changes to the NSSS design transients with respect to temperature, pressures, and frequency of occurrence. The acceptance criteria for the RCS piping and supports entail limiting the stresses to ASME Code Section III allowable stresses, and limiting the fatigue usage factors to less than 1.00.

The 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, although not bounding for all PWRs, were conservatively selected to be bounding for candidate plants with a high degree of confidence. The staff agrees with the DOE assessment that there is no effect from the TPC on the RCS piping and support stress and fatigue analyses, because 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, because the representative plant has a high power rating and, therefore, conservative plant parameters.

However, for specific plant implementation, statements of impact for the LOCA evaluations, RCS parameters, and NSSS design transients need to be addressed by licensees participating in DOE's program for the CLWR production of tritium in order to determine the impact for the RCS piping and supports. The staff agrees with DOE's assessment.

Reactor Coolant Pumps

The reactor coolant pump (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 effect of using TPBARs was evaluated by examining the changes in the key input parameters for the structural analysis, and then determining the effect on the analysis.

RCPs are subject to cold-leg transients. The design transients have been determined to be unaffected. Typical LOCA forces would not increase by more than 0.2 percent, as described in Section 2.3.3.1 of the TPC topical report. 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.

DOE 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.

DOE concludes that the RCP hydraulics and motor function is acceptable for the TPC because 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 is minimal and the RCP impact on its performance is acceptable. The staff agrees with DOE's assessment.

Auxiliary Heat Exchangers, Tanks, Pumps, and Valves

One 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.

Consistent with the function of all safety-related equipment and components, the components identified above are designed to fulfill 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 by DOE. The only transients that could be affected 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 of the TPC topical report) 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.

On this basis, no input parameters or assumptions have changed from the analysis of record as a result of the TPC. Therefore, the analyses of record remain valid.

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

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 key inputs included reactor vessel normal operating temperatures, NSSS design transients, and reactor vessel/reactor internals interface loads. Information concerning how each of the key inputs would vary with the implementation of the TPC was evaluated by DOE. The evaluations of the information on the individual inputs are summarized in the paragraphs that follow.

The reference plant reactor vessel is analyzed for an operating plant temperature differential that envelops the vessel operating temperature differential for the TPBAR program. No additional reactor vessel thermal and structural analyses are warranted by the changes in operating temperatures. In addition, the NSSS design transients are unchanged as a result of the full-core TPBAR implementation. Therefore, DOE concludes that 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. Therefore, DOE concludes that no additional structural analysis is needed 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). On the basis of DOE's evaluation results, no revisions to the reactor vessel stress report for the reference plant are considered necessary for TPC implementation.

For the above reasons, DOE concludes that 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 differing 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 addressed by licensees participating in DOE's program for the CLWR production of tritium to determine whether there is any effect on the reactor vessel structural evaluation. The staff agrees with DOE's assessment.

2.3.5 Control Rod Drive Mechanism (SRP Section 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 effect of the TPC was evaluated by examining the changes in the key input parameters for the structural analysis, and then determining the effect on the analysis.

The CRDMs are subject to cold-leg transients. The reference plant design transients apply without change to the proposed design modification as discussed earlier. Typical LOCA forces would not increase by more than 0.2 percent, as described in Section 2.3.3.1 of the TPC topical report. 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 of the TPC topical report 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 that limit. Because the reference plant design transients do not change, DOE concludes that the generic CRDM analysis continues to apply. LOCA and seismic loading conditions are resolved in concert with the reactor pressure vessel system analysis. In this analysis, site-specific seismic loads were used in the dynamic model for the reference plant to determine the loading on the CRDM. The applied CRDM loads were then compared to the allowable generic loads and found to be acceptable.

Therefore, the staff agrees with the assessment that the CRDMs satisfy the ASME Boiler and Pressure Vessel Code requirements based on the justification provided by the generic stress report, the analysis of record, and the LOCA and seismic load evaluations.

2.3.6 Reactor Internals (SRP Section 3.9.5) Evaluation

The effect of the TPBARs on the seismic response of the fuel assemblies is discussed in Section 2.4.2 of the TPC topical report. The related sections of the SRP with respect to the fuel assembly integrity have been reviewed to consider those factors where the TPBARs 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 required analyses.

The normal tritium production core (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. Source rods and TPBARs were combined on the same base plate 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. Some hardware changes will be required to facilitate the handling of the TPBARs and source rods in the spent fuel pool and provide a means of combining the secondary source rod with unirradiated TPBARs.

A discussion relating to the impact of the additional weight of each fuel assembly on the fuel assembly structural analysis and the grid load margin available for the reference plant appears in Section 2.4.2 of the TPC topical report. The structural adequacy of the Westinghouse fuel assembly design is evaluated using the methodology identified in Appendix A to SRP Section 4.2 for combined seismic and LOCA loads. The grid load results for the 17x17 VANTAGE+ fuel assembly design in the reference plant are discussed in the TPC topical report. The combined seismic and LOCA grid load is determined to be less than the allowable grid strength with adequate grid load margin available. Because the fuel 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 a part of the fuel assembly nozzle support. The added TPBAR assembly weight, together with the rodlet stiffness, has a negligible effect on the fuel assembly's dynamic characteristics. Therefore, in plants that will use the TPBAR assembly, DOE concludes that there will be an insignificant effect on the fuel assembly structural integrity evaluation.

The TPBARs can be treated as part of the fuel assembly nozzle support since they are within the guide tubes. 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. As a result, the dynamic characteristics of the fuel assembly are not affected. These factors reduce the effect of the added weight of the TPBAR assemblies on the LOCA/seismic analysis for the reference plant.

With regard to flow-induced vibrations, the response of a given set of reactor vessel internals is generally influenced by (1) hydraulic design parameters such as flow rates and vessel/core inlet and outlet temperatures, and (2) 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 earlier, the TPBARs have an insignificant effect on the dynamic characteristics of the fuel assembly. Consequently, the staff agrees with the assessment in the TPC topical report that the structural integrity of the reactor vessel internals with the TPC will not be adversely impacted with regard to LOCA and seismic loads as well as flow-induced vibrations.

There are, however, some potential analytical issues related to the reactor internals that would need to be considered for plant-specific implementation. These issues are summarized below:

The reference plant reactor internals were not designed to the ASME Code. To incorporate TPBARs in a plant whose internals are designed per the ASME Code, additional structural evaluation effort would be required to verify conformance with ASME design limits.

The reference plant does not have a thermal shield system. To incorporate TPBARs in a reactor with a thermal shield system, 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 previously qualified fuel cycles do not envelop 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 existing grid load margin for the reference plant (as discussed above), no LOCA/seismic analysis was deemed necessary. For a plant with significantly less grid margin, a plant-specific evaluation may be required to include the LOCA/seismic analysis. This would involve the development of a plant-specific fuel assembly dynamic model to establish structural input for the LOCA structural analysis.

Therefore, for plant-specific implementation, DOE concludes, and the staff agrees, that the technical matters listed above (i.e., ASME Code internals, thermal shield system, and grid load margin) should be addressed by licensees participating in DOE's program for the CLWR production of tritium to determine their effect on the reactor internals.

2.3.7 Equipment Qualification (SRP Section 3.11) Evaluation

DOE performed an evaluation to determine the effect of the TPC on environmental qualification (EQ) of electrical equipment in the reference plant. The major effect on EQ for a TPC plant, as compared with a plant with conventional core design, is the potential change in the radiation dose and dose rates resulting from differences in the core radiation sources. The other major effect on EQ for a TPC plant is the potential change in the mass and energy release for postulated LOCAs and for postulated secondary-system pipe ruptures.

<u>Potential Change in Radiation Dose And Dose Rates</u>

DOE assessed the effect of the TPC on the equipment qualification doses for several different accident and normal operating conditions, such as post-LOCA, high-energy line break, and normal operating doses for both inside and outside the containment. These conditions are consistent with SRP Section 3.11 and the Westinghouse report WCAP-8587. In WCAP-8587, Westinghouse outlined the methodology it used to qualify NSSS safety-related electrical equipment within its scope of responsibility and subject to a harsh environment.

In Section 2.3.7 of the TPC topical report, DOE states that, in general, the dose rate associated with the longer lived nuclides are expected to be lower for a TPC than for current operating plants since the TPC fuel assembly discharge burnups are lower than in conventional fuel cycles. DOE demonstrated that the doses considered in WCAP-8587 exceed the values associated with the TPC design.

The calculated gamma dose rates 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. The TPC plant dose rates are lower than the WCAP-8587 values by factors ranging from 1.13 to 2.24. The results are even more conservative for the reference plant, which has a dry containment. Specifically, the free volumes of a dry containment are higher than the free volume in ice-condenser containments by more than a factor of 2 and, because the sources and doses are inversely proportional to containment volume, the reference plant (dry containment) TPC doses would be lower by at least a factor of 2. An additional conservatism in the analysis is the fact that no credit is taken for shielding that is provided by internal structures such as shield walls and equipment.

The calculated beta dose rates as presented in WCAP-8587 are compared to those calculated for the plant operating with a TPC. The TPC doses, including the beta doses from tritium, conservatively assuming that the total end-of-life tritium activity inventory is distributed within the containment atmosphere, are noted to differ from the WCAP-8587 dose rate contribution values by factors ranging from 0.99 to 1.38. At short times after a LOCA, the tritium dose rate contribution is lower than the WCAP-8587 dose rate contribution by at least two orders of magnitude due to fission products. Because of the relatively long half-life (i.e., 12.3 years) of tritium, the relative dose rate contribution increases so that the WCAP and TPC values are essentially the same at 1 year after a LOCA. The TPC total integrated doses at the post-LOCA time period of interest (i.e., less than 1 year) will not exceed the bounding doses contained in WCAP-8587.

On this basis, the staff concludes that the dose rates and integrated doses considered in the WCAP-8587 report bound the values associated with the TPC design. Therefore, the incorporation of TPC design does not have any effect on the EQ radiation dose and dose rates.

Potential Change in the Mass and Energy Release for Postulated Loss-of-Coolant Accidents

DOE assessed the effect of the TPC on EQ due to the potential change in mass and energy release for a postulated LOCA. The mass and energy releases in LOCA events are considered for short-term subcompartment analysis and long-term containment integrity analysis. The short-term subcompartment analysis is concerned with the maximum differential pressure 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.

The evaluation involves a comparison of key safety analysis parameters used in the typical Westinghouse LOCA mass and energy analysis process for a four-loop plant design for the short-term and long-term transients. Following a qualitative assessment of these key parameters, a comparison is made with respect to the analyses of record being evaluated. The key parameters for both the short-term and long-term LOCA releases are the RCS conditions and the core-stored energy for the long-term analysis only. Because the RCS conditions for the TPC are identical to the conditions for the reference plant, DOE concludes that there would not be any change in the reference plant's short- or long-term LOCA mass and energy releases. An evaluation of the effect of a full complement of TPBARs showed a negligible affect 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 the TPC topical report provides a comparison of core and vessel pressure drops, and shows that these pressure drops do change slightly (i.e., less than 3 percent) for normal operation. The pressure drop changes would not have an effect 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 the TPC topical report provides information for the minimum containment back pressure calculations for determining the LOCA peak cladding temperature. This evaluation determined that the presence of TPBARs could result in a decrease in the initial RCS inventory. A decrease in inventory would be a benefit for the LOCA mass and energy release for containment integrity analysis.

On the basis of the conclusion that none of the key analysis parameters are adversely affected 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 and, therefore, within the design limit and equipment qualification profiles. However, if the design operating parameters are affected for a specific plant implementation, a plant-specific EQ evaluation is required.

Potential Change in the Mass and Energy Release for Postulated Secondary System Pipe Rupture

DOE also assessed the effect of the TPC on EQ from potential change in steamline and feedline break mass energy release. Mass and energy releases in non-LOCA events due to high-energy secondary-side line breaks (steamline and feedline breaks) are sensitive to changes in core reactivity coefficients; specifically, moderate 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.

The DOE assessment involves a comparison of key safety analysis parameters used in the standard Westinghouse reload safety evaluation process. Following a qualitative assessment of these key parameters, a comparison was made with respect to the high-energy secondary-side line break analyses of record being evaluated. As discussed in Section 2.4.3 of this report, all of the key safety analysis parameters that constitute the reference plant safety analyses (for steamline break and feedline break mass and energy releases) bound the core reload design values with TPBARs.

On the basis of the above conclusion that none of the key safety analysis parameters has been exceeded for the reference plant core reload design with TPBARs, the licensing basis of record for the high-energy secondary-side line breaks continue to remain valid. However, the staff concludes that a plant-specific EQ evaluation will need to be undertaken by licensees participating in DOE's program for the CLWR production of tritium if the design operating parameters are affected for a specific plant implementation.

Conclusion

On the basis of its review, the staff concludes that the incorporation of TPBARs will not have a significant effect on EQ of safety-related electrical equipment and will not cause the post-accident environment to exceed the EQ envelope and, therefore, is acceptable. However, if the design operating parameters are affected for a specific plant implementation, a plant-specific EQ evaluation is required.

2.4 Reactor

2.4.1 Introduction

Section 2.4 of the TPC topical report deals 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.

2.4.2 Fuel Design (SRP Section 4.2) Evaluation

The fuel design was reviewed in accordance with the guidelines in Section 4.2 of the SRP. The objectives of this fuel system safety review are to provide assurance that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs), (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained.

The fuel design parameters for the TPC are essentially the same as a standard Westinghouse VANTAGE+ fuel assembly containing a 17x17 fuel rod array with intermediate flow mixer (IFM) grids and Zirlo cladding. The primary difference between the TPC design and typical reload cores is the use of TPBARs, which will be inserted into the 24-rod cluster control assembly (RCCA) channels in most of the fuel assemblies that do not actually contain RCCAs. 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 core reactivity. A more complete design description is given in Section 3.2.1 of the TPC topical report. A few TPBAR assemblies will consist of TPBARs and either primary or secondary source rods. To facilitate removal and reinsertion of the source rods from the TPBAR base plate assemblies, a design change to the base plates will probably be necessary. If that is so, the specifics of the revised design must be developed once a specific plant is selected and the production requirements are defined.

In addition to the high loading of TPBARs, large numbers of Westinghouse integral fuel burnable absorber (IFBA) fuel rods will be used. IFBA rods consist of a thin coating of ZrB_2 on the fuel pellet, which serves to reduce the initial core reactivity and shape the power distribution. This type of rod has been used in many Westinghouse plants.

DOE's assessment comparing the effect of the additional weight of each fuel assembly due to the TPBARs to the grid load margin available for the reference plant indicates that sufficient grid load margin would still exist. Therefore, the staff concludes that the use of TPBAR assemblies in the reference plant has no adverse impact on the structural integrity of the fuel assembly.

Fuel rod design criteria have been established for all normal operating conditions and AOOs. These design criteria are rod internal pressure, cladding stress, cladding strain, cladding oxidation and hydriding, fuel temperature, cladding fatigue, cladding flattening, and fuel rod axial growth. In addition, the NRC has approved the methodology used in the evaluation of fuel rod design criteria for design applications up to a lead rod average burnup of 60,000 MWD/MTU (WCAP-10125-P-A). The NRC-approved PAD 3.4 computer code, with NRC-approved models for in-reactor behavior, was used to calculate the fuel rod performance over its irradiation history (WCAP-11873-A, WCAP-12610-P-A).

Each of these criteria has been verified for the TPC fuel in the reference plant by evaluating the predicted performance of the limiting fuel rod, defined as the rod that gives the minimum margin to the design limit. On the basis of the assessments presented in the topical report, the staff concludes that the fuel rod design criteria are satisfied for the TPC reference core. However, although these analyses have shown that margin exists to the design criteria limits for the reference core, DOE concludes, and the staff agrees, that implementation in specific plants should be addressed by licensees participating in DOE's program for the CLWR production of tritium to determine acceptability.

2.4.3 Nuclear Design (SRP Section 4.3) Evaluation

The staff based its review of the nuclear design on information in the topical report and the applicant's responses to staff requests for additional information. The staff conducted its review in accordance with the guidelines in SRP Section 4.3. The objective of this review is to ensure the acceptability of the core power distribution, the reactivity coefficients, the control requirements, and the analytical methods. Also reviewed were the criticality calculations for single assemblies and groups of assemblies as related to fuel storage considerations.

A representative Westinghouse four-loop plant was used as the reference reactor. The cycle energy chosen was 494 effective full power days (EFPDs), which is equivalent to an 18-month cycle at a capacity factor of approximately 90 percent, and is representative of a typical PWR fuel cycle. The corresponding cycle burnup is 21,564 MWD/MTU.

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

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

The tritium product (T) has a 12.3-year half-life and decays into helium-3. Therefore, a secondary neutron absorption reaction is

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

The lithium-aluminate pellet also contains a large amount of ^{7}Li. Consequently, ^{6}Li, ^{7}Li, ^{3}He, and T neutron cross sections are required. Since none of these isotopes are in the standard PHOENIX-P library, cross section data for them were added using the basic nuclear data from ENDF/B-VI (BNL-NCS-17541) and new versions of the NRC-approved PHOENIX-P and ANC codes (WCAP-11596-P-A), called PHOENIX-L and ANC-l, were developed. PHOENIX-L calculates the fuel and burnable absorber cross-section data for ANC-L, which uses these data to predict core power distributions, reactivity coefficients, etc. The NRC reviewed the revised codes and concluded that they were adequately verified (NRC letter dated September 15, 1997). In addition to the PHOENIX-L and ANC-L codes, the APOLLO code (WCAP-13524-P-A) was used for some of the nuclear design analyses. Since ANC-L provides the necessary information for APOLLO to model TPBARs, no modifications to the standard version of APOLLO are required.

Enrichment zoning is employed in the two types of fresh fuel assemblies in the TPC, which Westinghouse calls "high enrichment" and "low enrichment" fuel. For the high enrichment fresh fuel, 68 fuel rods in the outer row and an additional rod in each corner of the assembly are enriched to 4.15 weight percent ^{235}U. The remaining 196 rods are enriched to 4.95 weight percent ^{235}U. The low-enrichment fresh fuel assemblies contain 68 rods enriched to 3.45 weight percent ^{235}U and 196 rods enriched to 4.60 weight percent ^{235}U. The fresh fuel assemblies in the reference core contain uniformly enriched rods, 4.20 weight percent in the high-enrichment fuel, and 4.00 weight percent ^{235}U in the low-enrichment fuel. In addition to the slightly higher enrichments, larger feed regions are used in the TPC design compared to typical current designs. The feed region size of the equilibrium cycle is 140 assemblies as compared to 84 – 92 assemblies typically used in a four-loop plant.

<u>Fuel Burnup</u>

The initial excess reactivity of the TPC design is larger than for the reference plant in order to compensate for the residual reactivity penalty of the TPBARs at end of life (EOL). This initial excess reactivity is controlled by the soluble boron in the RCS, the IFBA rods, and by the TPBARs themselves. The fuel average discharge burnup for the TPC designs (less than 30,000 MWD/MTU) is smaller than typical for the reference plant (about 45,000 MWD/MTU) because of the large feed region size.

Reactivity Coefficients

The reactivity coefficients are expressions of the effect on core reactivity of changes in such core conditions as power, fuel and moderator temperature, moderator density, and boron concentration. These coefficients vary with fuel burnup and power level. The staff reviewed the calculated values of reactivity coefficients and concludes that they adequately represent the full range of expected values. The staff reviewed the reactivity coefficients used in the transient and accident analyses and concludes that they conservatively bound the expected values, including uncertainties. Further, moderator and power Doppler coefficients along with boron worth are measured as part of the startup physics testing to ensure that actual values are within those used in these analyses.

Although the fuel temperature (Doppler) coefficient for the TPC is always negative, it is slightly less negative than the values for the reference core at beginning of life (BOL). Therefore, those accidents that are sensitive to this parameter (uncontrolled RCCA bank withdrawal from subcritical, RCCA ejection at BOL, and hot zero power excessive feedwater flow) have been reevaluated as discussed in Section 2.15 of this report.

The moderator temperature coefficient (MTC) for the TPC is slightly more negative at BOL than that for the reference plant because of the lower soluble boron concentration. This results in a less-severe anticipated transient without scram (ATWS) event, as discussed in Section 2.15.7 of this report. At end-of-life (EOL), the MTC is slightly less negative. An MTC of up to +7.0 pcm/^{o}F is allowed from 0 percent to 70 percent power. From 70 percent to 100 percent power, the MTC decreases linearly from +7.0 to 0.0 pcm/^{o}F.

The total power coefficient for the TPC design is always negative at all power levels and times in cycle. Therefore, the reactivity feedback behavior of the TPC design meets General Design Criteria (GDC) 11, which states that the design should be such that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Power Distribution

The design basis for control of power distributions is that, with at least a 95-percent confidence level

- The fuel will not be operated at more than 14.5 kW/ft under normal operating conditions, including an allowance of 2 percent for calorimetric error in order to meet the initial conditions assumed in the LOCA analysis.

- Under abnormal conditions (including maximum overpower), the peak fuel power will not produce fuel centerline melting.

- The core will not operate during normal operation or AOOs with a power distribution that will cause the departure from nucleate boiling ratio (DNBR) to be less than the design limit DNBR for the reference plant (i.e., 1.24 using the W-3 DNB correlation).

Power distributions representative of the TPC designs were calculated using the NRC-approved methods presented in (WCAP-9272-P-A) and are shown there in Figures 2.4.3-11 through 2.4.3-26. Hot full-power (HFP) assembly radial power distributions as a function of burnup (BOL no xenon, BOL equilibrium xenon, middle of life [MOL], and EOL) for first cycle and equilibrium cycle are presented for AOO conditions and for control bank D at its HFP insertion limit of 161 steps. BOL and EOL fuel rod power are also shown for first cycle. HFP BOL and EOL axial power shapes for first and equilibrium cycles are compared with the reference core.

In response to a staff request for additional information (DOE letter dated December 2, 1998), DOE has shown that, in general, the effect of ^{3}He buildup following an extended shutdown of 6 months near EOL would slightly decrease the power in the interior core locations while slightly increasing the power in the outer rows of fuel assemblies near the core periphery. Overall, the core best estimate $F_{\Delta H}$ decreases from 1.297 to 1.288 due to the ^{3}He buildup. The core maximum F_Q was found to increase slightly from 1.420 to 1.437 due to the ^{3}He buildup. These peaking factor effects are small enough to be accommodated within existing peaking factor margins.

The effects of flux peaking caused by axial gaps between absorber pellets in a pellet stack or between pellets in adjacent pencils is evaluated in Section 3.7.2 of the TPC topical report. TPBAR absorber pellets are contained in "pencils," which are stacked in a column in the TPBAR. The interfaces between the pencils result in gaps between segments of absorber pellet material. Each gap produces a small local axial power peak in the adjacent fuel rods. Gaps are affected by manufacturing tolerances, temperature, and irradiation. A full three-dimensional evaluation of the power peaking in the fuel rods adjacent to TPBARs with worst-case gaps resulted in peaking of less than 3 percent for burnups below 10,000 MWD/MTU and less than 5 percent for burnups above 10,000 MWD/MTU. Because the rod powers adjacent to the TPBARs are generally depressed by at least this amount, the peaking effects due to the TPBAR gaps, which affect only about a 1-inch height of the neighboring fuel rods, can be readily accommodated.

The predicted power distributions demonstrate that these reference plant power peaking factor limits and DNB design bases are met for the TPC design. NRC-approved methods were used in these calculations. In addition, evaluation of the overpower transients discussed in Section 2.15 of this report demonstrate that the power does not exceed the 22.4 kW/ft limit, thereby ensuring that fuel melting is precluded during AOOs. However, any future changes to the TPBAR design, the number of fuel assemblies containing TPBARs, and/or the type of CLWR used to host the TPBARs will require a reevaluation of the acceptability of the resulting power distributions.

<u>Control Requirements</u>

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission-product buildup, a significant amount of excess reactivity is built into the core. DOE has provided sufficient information relating to core reactivity balance for the first and equilibrium cycle cores, and has shown that means are incorporated into the design to control excess reactivity at all times.

Soluble boron is used to compensate for slow reactivity changes, including changes associated with fuel burnup, changes in xenon and samarium concentration, buildup of longlife fission products, burnable poison rod depletion, and the large moderator temperature change from cold shutdown to hot standby.

The staff reviewed the calculated rod worths at BOL and EOL for the TPC designs. The staff concludes that DOE's assessment of reactivity control is suitably conservative, and that adequate negative reactivity worth is provided by the control system to ensure that the TPC will meet the 1.3 percent $\Delta\rho$ shutdown margin requirement for Modes 1 and 2, assuming that the most reactive RCCA is stuck in the fully withdrawn position.

Except for their production and retention of tritium, the TPBARs act much the same as the conventional burnable poison rod assemblies (BPRAs) under normal operation. Because the TPBARs burn up at a slower rate than conventional discrete burnable absorbers, they are better for reactivity control. However, failure of the TPBARs must be considered in order to determine whether they would continue to provide an appropriate level of reactivity to maintain the reactor in a safe state, in conjunction with the soluble boron and control rods. Multiple, widespread TPBAR failures are considered in the LBLOCA analysis. DOE has indicated that testing and analyses demonstrate that only a small portion of the ^{6}Li is lost when TPBARs fail. Therefore, because the ^{6}Li is the only neutron absorbing component of the TPBARs, even for failures involving a breach of the cladding, the TPBARs would continue to maintain sufficient ^{6}Li to control core reactivity.

Stability

DOE discusses the stability of the reactor to xenon-induced power distribution oscillations, and the control of such transients. Because of the negative power coefficient, the reactor is inherently stable to oscillations in total reactor power.

The core may be unstable to axial xenon oscillations. However, as for the reference plant, axial oscillations will be detected and controlled before any safety limits are reached, thus preventing any fuel damage. DOE also concluded that the core will be stable to both radial and azimuthal xenon oscillations throughout core life based on the similarity of the TPC reactivity feedback with that of the reference core. The staff agrees with this conclusion.

2.4.4 Thermal and Hydraulic Design (SRP Section 4.4) Evaluation

The thermal-hydraulic aspects of the TPC were analyzed by Westinghouse for DOE in accordance with the acceptance criteria outlined in Section 4.4 of the SRP. The principal thermal-hydraulic-induced design basis for the TPC is the avoidance of thermal-hydraulic-induced damage during normal steady-state operation and anticipated operational transients. DOE performed a thermal-hydraulic analysis to ensure the safe operation and integrity of the core and fuel assemblies for steady-state operation and transients categorized under Condition I and II events (DOE letter dated July 30, 1998).

The methodology utilized to determine the acceptability of the TPBARs is consistent with the current standard Westinghouse methods for inserting new components into Westinghouse cores. In the analysis, DOE assumed that the TPBAR is a core component not unlike other core components that are routinely inserted into the core for purposes of reactivity and power distribution control.   Specifically, the TPBAR is essentially just another (different) type of burnable absorber. The physical design of the TPBAR is very similar to the physical design (outside geometry) of the existing burnable absorber rods.

DOE performed an evaluation to determine the effects of the representative reactor core thermal-hydraulic conditions on the function of the integrity of the TPBARs. Utilizing a Westinghouse reference plant [a typical four-loop plant loaded with 17x17 VANTAGE+ fuel assemblies and with intermediate flow mixers (IFMs)], standard Westinghouse procedures were 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 thimble guide tubes.

A calculation was performed utilizing fuel data, TPBAR data, and appropriate core limits for the reference core with the THINC-IV thermal-hydraulic computer code. The result of the calculation was used to define core conditions, such as core and assembly flow and channel enthalpy rise. These conditions are then used as boundary conditions for the response of the core component (WCAP-7956, WCAP-8054). The THINC-IV computer code was used to determine the flow distribution in the core and the local conditions in the hot channel for use in the DNB correlation.

Assembly flow is broken down into “channel flow” and “thimble flow.” The “channel flow” information is obtained by running THINC-IV with the reference core channel input data and boundary conditions. The “thimble flow” analysis is a calculation of the flow through the fuel assembly thimble with the TPBAR in the thimble hole. Flow information is determined by analysis with the methodology that is used in the standard Westinghouse thermal-hydraulic design procedure.

To determine the thimble tube flow, the methodology solves the continuity, momentum, and energy equations for a flow system consisting of two parallel paths: the thimble tube path and the channel flow path (flow outside the thimble tube). The channel flow path fluid conditions are known from performing a THINC-IV calculation, thus simplifying the problem to that of flow in a single channel in which the boundary conditions are specified. The process takes the core pressure drop and performs an iterative calculation, on a node-by-node basis, up the absorber rod in an effort to match the pressure drop from the “channel” side to that of the flow through the thimble itself. The analysis of the flow through the thimble incorporates absorber heat rates, internal material characteristics, and assembly power peaking. The output of the calculation(s) results in obtaining absorber temperature, thimble flow, and local pressure, as well as thimble and TPBAR cladding temperature behavior.

In analyzing the inclusion of the TPBAR into the standard 17x17 fuel assembly, DOE performed two analyses: (1) An axial shape study, and (2) a rod withdrawal from subcritical transient analysis.

In the axial shape study, DOE analyzed the inclusion of the TPBAR in the assembly to ensure that its presence did not result in a more limiting power distribution. The analysis consisted of comparing axial power shapes from depletion runs of the TPBAR assembly to those of the reference core axial power shapes depletion runs. These comparisons formed the basis for the thermal-hydraulic design analyses. Results of the analysis indicated that the incorporation of the TPBAR does not result in a more limiting power distribution.

DOE analyzed the rod withdrawal accident to demonstrate that the DNBR for the TPC inserted core continues to be acceptable. The results of the analyses showed that the limiting acceptance criterion for this event, which is to demonstrate that the reactor protection system is adequate to preclude a violation of the DNB design basis, is still satisfied. DOE also conducted studies to ensure that the mechanical integrity of the thimble and the TPBAR component remain functional during insertion or removal, and that the structural integrity remains sound and serviceable.

To prevent excessive heat and corrosion, a design criterion was imposed on the TPC so that the TPBAR component did not exceed its melting temperature. This criterion will prevent surface boiling from the core component within the dashpot region of the thimble. The imposed design criterion will not permit bulk boiling along the length of the thimble and the sum of the flow through all the thimble, and the TPBAR combinations must be less than that allowed by the bypass flow limits that are used to ensure adequate flow for core cooling.

To ensure the acceptability of the TPC, DOE conducted analyses assuming insertion of the TPBARs in a core with both types of fuel in the reference core: (1) VANTAGE+ fuel assemblies with IFMs, and (2) VANTAGE+ fuel assemblies without IFMs. The function of the IFMs is to enhance mixing and improve cooling, allowing for greater peaking factors in the core power distributions.

Analyses of relaxed axial offset control comparisons between the TPBAR core and the reference core showed that the power shapes and analysis for the reference core would remain bounding. The introduction of the TPBAR component did not disturb the power profile beyond those that are already bounded within the thermal-hydraulic design bases.

<u>Conclusion</u>

DOE utilized standard analytical methods 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 specified criteria. These criteria are the same criteria used to analyze typical Westinghouse fuel with burnable poison rod assemblies. That is, the thermal-hydraulic design basis of the TPBARs will continue to ensure that the TPBARs will meet the requirements of Section 3.6.2 of the TCP topical report and are below the operating temperature analyzed in the structural analysis of Section 3.4 of the TCP topical report.

DOE’s analyses regarding the incorporation of the TPBARs in the reference plant showed that the bypass flow will remain within its design limit of 8.4 percent, and that the DNB criterion will continue to be met with no feature of the TPBAR component affecting the coolability of the core. The staff agrees with this assessment. However, the continued compliance with the DNB criterion, given the operating conditions of a particular plant, must be evaluated. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

2.4.5 RCCA Drop Time (SRP Section 4.6) Evaluation

During full-power operation, all RCCAs are fully (or near fully) withdrawn from the core. If any event necessitates a reactor trip, the RCCAs are released and drop to their fully inserted position under gravity. It is anticipated that the addition of TPBARs will slightly increase the core pressure drop, thus increasing the RCCA drop time compared to the reference plant without TPBARs.

The RCCA drop time acceptance criterion for the reference plant is defined in the technical specifications (TS) as 2.7 seconds from the initiation of the drop to the time it reaches the top of the dashpot. RCCA drop times were calculated for various combinations of plant operating and external force conditions to determine whether, with TPBARs, the RCCAs will drop under the TS time of 2.7 seconds. On the basis of these calculations, DOE concludes, and the staff agrees, that the overall effect of adding the TPBARs was insignificant and the RCCA drop times were all less than the reference plant TS limit of 2.7 seconds.

2.5 Reactor Coolant System and Connected Systems

2.5.1 Introduction

The sections in Chapter 5 of the SRP identify criteria to review the adequacy of systems that maintain the integrity of the reactor coolant pressure boundary. DOE's assessment of the effect of TPBARs on overpressurization protection, reactor vessel materials and pressure-temperature limits, and the residual heat removal system are discussed in Sections 2.5.2 through 2.5.4 below.

2.5.2 Cold Overpressure Mitigation (SRP Section 5.2.2) Evaluation

The cold overpressure mitigating system (COMS) is designed to prevent violations of the limits of Appendix G to 10 CFR Part 50 during low temperature operating conditions. This system is known in the industry as the "low temperature overpressure protection system" (LTOPS). On the reference plant, the pressurizer power-operated relief valves (PORVs) and/or the residual heat removal (RHR) relief valves are used to protect the pressure/temperature limits during an overpressure transient caused either by a mass addition or heat addition to the reactor coolant system.

DOE performed an evaluation to determine the effects of the incorporation of a full complement of TPBARs to the COMS. DOE concluded that the effect of the TPBARs is to lower the Appendix G limits as a result of the changes in the neutron fluence on the reactor vessel. For the reference plant, these pressure/temperature limits will be reduced by approximately 5 psi or less over the pressure range of protection. (For older plants with RCS pressure/temperature limits and COMS setpoints specified in TS, this slight change could require an amendment to the TS.) No changes to thermal/hydraulic plant behavior are expected. Therefore, DOE 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 relief valve performance and flow capacities would not be affected.

The staff agrees with DOE's conclusions.   However, for any plant for which the Appendix G to 10 CFR Part 50 limits change, COMS setpoints would need to be revised using the latest NRC-approved methodology documented in WCAP-14040-NP-A.

2.5.3 Reactor Vessel Integrity (SRP Sections 5.3.1, 5.3.2, and 5.3.3) Evaluation

Introduction

The requirements for maintaining reactor vessel integrity are contained in Appendices G and H to 10 CFR Part 50 and in 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock (PTS)." Appendix G to 10 CFR Part 50 specifies pressure-temperature (P-T) limits for operation of the reactor vessel and requirements for the Charpy upper-shelf energy (USE). Appendix G to 10 CFR Part 50 requires that the initial unirradiated USE at the start of the vessel life be no less than 102 joules (75 ft-lb), and that the vessel maintain a USE level no less than 68 joules (50 ft-lb) throughout the service life. If it is anticipated that a vessel might fall below 68 joules (50 ft-lb) before license expiration, an analysis must be submitted that demonstrates “margins of safety against fracture equivalent to those required by Appendix G of the ASME Code.” This analysis is subject to the approval of the director of the Office of Nuclear Reactor Regulation.

Appendix H to 10 CFR Part 50 specifies material surveillance program requirements to monitor the changes in the fracture toughness properties in the reactor vessel beltline region of light-water nuclear power reactors that result from exposure of these materials to neutron irradiation and the thermal environment. Appendix H requires that the surveillance program and capsule withdrawal schedule meet the requirements of the edition of American Society for Testing and Materials (ASTM) E 185 that was current on the issue date of the ASME Code to which the reactor vessel was purchased. Later editions of ASTM E 185 may be used, but only those editions through 1982.

The PTS rule (10 CFR 50.61) establishes screening criteria that define a limiting level of embrittlement beyond which operation cannot continue without further plant-specific evaluation. The screening criteria are given in terms of reference temperature RT_{PTS} . The PTS rule establishes a methodology for calculating the RT_{PTS} for reactor vessel beltline plates, forgings, and welds.

<u>Evaluation</u>

The TPC topical report identifies the applicable regulations and describes methods for demonstrating compliance with Appendices G and H to 10 CFR Part 50 and to 10 CFR 50.61. The topical report contains material test and neutron fluence data for a reference plant. The neutron fluences are estimated with and without TPBARs at the project end of life, 32 effective full-power years (EFPYs). The neutron fluence without using tritium production cores at 32 EFPYs was estimated as 2.03×10^{19} n/cm^2 (E>1.0 Mev) from surveillance test results from the reference plant. If a TPC were inserted at the end of the fourth refueling outage in the reference plant and continued to be used, the neutron fluence at 32 EFPYs would increase to 2.06×10^{19} n/cm^2 (E>1.0 Mev). These data are used to calculate P-T limits and RT_{PTS} values at 32 EFPYs for the reference plant. The RT_{PTS} value for the reference plant was calculated to be less than the screening criteria in the PTS rule. The RT_{PTS} value for the reference plant increased from 126 ^{o}F without TPCs to 127 ^{o}F with TPCs.

The TPC topical report identifies a surveillance program that complies with ASTM E 185-82.

In its letter dated December 2, 1998, DOE submitted the results of an equivalent margin analysis (EMA) for Watts Bar to demonstrate that its reactor vessel would meet the safety margins of Appendix G of the ASME Code. Because the NSSS design is essentially identical at Watts Bar and Sequoyah 1 and 2, this analysis applies to all three plants. DOE had a generic EMA for four-loop Westinghouse NSSS plants prepared for it by Westinghouse. The evaluation is documented in WCAP-13587, Revision 1. The generic evaluation was for reactor vessels with projected EOL USE of 43 ft-lb. The beltline material with the lowest USE at EOL in the reactor vessel is intermediate shell forging 05, which has a projected USE of 43.4 ft-lb using its chemistry and 46 ft-lb using the available surveillance data. Because the projected USE values are greater than the value used in the WCAP-13587 evaluation, DOE concludes that the reactor vessel would meet the safety margins of Appendix G of the ASME Code. However, this conclusion must be confirmed from irradiated material surveillance coupons from the Watts Bar and Sequoyah reactor vessel material surveillance programs.

Conclusion

The TPC topical report identifies the applicable regulations and describes methods for demonstrating compliance with Appendices G and H to 10 CFR Part 50 and with 10 CFR 50.61. In the TPC topical report, DOE concludes, and the staff agrees, that the reference plant's pressure/temperature limits report (PTLR) and final safety analysis report (FSAR) would need to be updated to reflect the change to the PTS value and include the updated P-T curves for the applicable EFPYs. In addition, because the reactor vessel integrity analyses are dependent upon the plant-specific materials properties and neutron fluence, the staff concludes that a licensee participating in DOE's program for the CLWR production of tritium must present the material properties for its reactor vessel and perform analyses that demonstrate it will meet the requirements of Appendices G and H to 10 CFR Part 50 and of 10 CFR 50.61. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

2.5.4 Residual Heat Removal System (SRP Section 5.4.7) Evaluation

DOE performed an evaluation to determine the effects of the incorporation of TPBARs on the design of the residual heat removal system (RHRS). DOE concluded that, because the decay heat from the reactor core will be reduced with TPBARs as a result of the reduced average maximum burnup of the core, the heat load will be reduced on the RHRS. Therefore, the design of the RHRS in the reference plant will not be negatively affected. The staff agrees with this assessment. However, the increase of the heat load in the spent fuel pool with TPBARs design would affect the component cooling water capacity and would indirectly affect the performance of RHRS. This effect is addressed in Section 2.9.3 of this report.

2.6 Engineered Safety Features

The sections in Chapter 6 of the SRP identify criteria to review the adequacy of the engineered safety features (ESFs) systems that are designed to protect the public in the event of an accidental release of radioactive fission products from the RCS. The DOE evaluations of the effect of the TPBARs on the mass and energy releases from LOCAs and secondary-system pipe ruptures, on the minimum containment pressure following a LOCA, and on the combustible gas control systems are described in Sections 2.6.2 through 2.6.5 of the TPC topical report.

The results of the staff's review of a TPC with a full complement of TPBARs in a currently licensed CLWR is summarized by SRP section. The reference plant evaluated in the TPC topical report was a newer-vintage, four-loop PWR with a large, dry containment.

Containment performance objectives are potentially affected by changes to the mass and energy releases to the containment from a spectrum (including break size and location) of postulated LOCAs (i.e., RCS pipe breaks) and secondary-system steam and feedwater line breaks. The TPC will reduce the mass of coolant in the primary system, being displaced by the TPC structures. This reduction in coolant mass (about 0.25 percent less than the reference core) also reduces the stored energy (about 0.27 percent less than the reference core) in the reactor coolant for release to the containment during a LOCA. The initial stored energy in the reactor core is estimated by DOE to increase slightly (less than 1 percent higher than the reference core) and the long term decay heat generation from gamma heating is estimated by DOE to be about 0.07 percent higher than the reference core. In addition, the TPC becomes a potential source for additional hydrogen for the combustible gas inventory control systems.

2.6.1 Control Room Habitability Systems (SRP Section 6.4)

DOE evaluated the effect of the TPBARS on control room habitability systems for the reference plant in accordance with Section 6.4 of the SRP. The acceptance criteria for the systems are based on meeting the relevant requirements of GDC 4, 5, and 19 of 10 CFR Part 50 in the control room, considering the use of the emergency ventilation system, the use of low-leakage dampers or valves, the ability of the ventilation system to pressurize the control room, the location of the control room with respect to the potential release points, the toxic gas hazard limits, and the radiation dose criteria. DOE states that only the radiation dose criteria are potentially affected by the incorporation of the TPBARs, and shows that, for the reference plant, the radiation dose criteria would be exceeded.

In its assessment, DOE did not include the whole body dose with the thyroid dose. The internal dose from tritium is a whole-body dose, as is documented in Federal Guidance Reports 11 and 12. This is not addressed explicitly in the SRP because, with the model source terms that have been used for typical power reactor operation, the dose from tritium is not significant. DOE suggested using the total effective dose equivalent (TEDE) criterion developed during the review of the designs of the advanced reactors, but this would have little significance for the tritium dose evaluation because the internal tritium dose is the TEDE (or whole-body equivalent dose). The staff concludes that the present SRP criteria are adequate and applicable, but additional control measures to control the dose from tritium will be needed if the reference plant were to be used as the tritium production facility.

As noted by DOE, the control room habitability concern is the direct consequence of the assumed high leak rate from the ECCS. The leak rate assumed (2 gpm) is the value formerly used as a default value for plants without a leakage reduction system. Because of the Three Mile Island “lessons learned” requirements, all plants now have leakage reduction programs. The ECCS leakage normally assumed in accident assessments is twice the leak rate that triggers corrective action under the applicable leakage reduction program; values of 2 gallons per hour or less are typically used. With such a low leak rate, the staff concludes that the reference plant would meet the relevant dose criterion. Thus, the control room habitability concern would not exist if the reference plant was assumed to have implemented a more stringent ECCS leakage reduction program.

Therefore, the staff concludes that, except for the dose criteria issue, the TPC topical report adequately addresses this matter, but that a plant-specific assessment will be needed. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to produce tritium for DOE.

2.6.2 Containment Functional Design

<u>PWR Dry Containments, Including Subatmospheric Containments</u>

The acceptance criteria used to evaluate PWR dry containments, including subatmospheric containments, are based on the relevant requirements of GDCs 13, 16, 38, 50, and 64 of Appendix A to 10 CFR Part 50. During its review of a nuclear power plant design, the staff evaluates the following containment design areas and containment performance analyses to verify that a design conforms to these requirements:

(1) temperature and pressure conditions in the containment from a spectrum (including

break size and location) of postulated LOCAs (i.e., RCS pipe breaks) and secondary-system steam and feedwater line breaks;

(2) maximum expected external pressure to which the containment may be subjected;

(3) minimum containment pressure that is used in analyses of emergency core cooling system capability;

(4) effectiveness of static and active heat removal mechanisms;

(5) pressure conditions within subcompartments that act on system components and supports due to high-energy line breaks;

(6) range and accuracy of instrumentation that is provided to monitor and record containment conditions during and following an accident.

In Sections 2.6.2 and 2.6.3 of the TPC topical report, DOE has shown that the TPBARs have no significant effect on the mass and energy releases for LOCA and secondary-system pipe ruptures. Therefore, DOE concludes, and the staff agrees, that the containment performance analyses and responses to LOCA or secondary system pipe ruptures are not expected to be affected by the TPC. DOE also concludes, and the staff agrees, that the TPC has no effect on the instrumentation needed to monitor and record containment conditions during and following an accident.

<u>Ice Condenser Containments</u>

The acceptance criteria used to evaluate ice condenser containments are based on the relevant requirements of GDCs 13, 16, 38, 39, 40, 50, and 64 of Appendix A to 10 CFR Part 50. During its review of a nuclear power plant design, the staff evaluates the following containment design areas and containment performance analyses to verify that a design conforms to these requirements:

(1) temperature and pressure conditions in the containment from a spectrum (including break size and location) of LOCAs (i.e., RCS pipe breaks) and steam and feedwater line breaks;

(2) The maximum expected external pressure to which the containment may be subjected;

(3) The design of the ice condenser system;

(4) The pressure conditions within containment internal structures that act on system components and supports due to high-energy line breaks;

(5) The maximum allowable operating deck steam bypass area for a full spectrum of reactor coolant system pipe breaks;

(6) The design provisions and proposed surveillance program to ensure that the ice condenser will remain operable for all plant operating conditions;

(7) The design of the return air fan systems;

(8) The effectiveness of static and active heat removal mechanisms;

(9) The minimum containment pressure that is used in the analyses of emergency core cooling system capability;

(10) The range and accuracy of instrumentation that is provided to monitor and record containment conditions during and following an accident.

The reference plant does not have an ice condenser containment. However, DOE concludes, and the staff agrees, that if the mass and energy releases do not change significantly with the incorporation of TPBARs, there will be no effect on the containment response of an ice condenser plant from the TPC.

Subcompartment Analysis (SRP 6.2.1.2)

The acceptance criteria for subcompartment analysis are based on the relevant requirements of GDCs 4 and 50 of Appendix A to 10 CFR Part 50. These requirements include consideration of the adequacy of the analysis used to determine the differential pressure values for containment subcompartments. A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high-energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within the volume. A short-term pressure pulse would exist inside a containment subcompartment following a pipe rupture within the volume. This pressure transient produces a pressure differential across the walls of the subcompartment that reaches a maximum value generally within the first second after blowdown begins. The magnitude of the peak value is a function of several parameters, which include the blowdown mass and energy release rates, the subcompartment volume, the vent area, and the vent flow behavior. A transient differential pressure response analysis should be provided for each subcompartment or group of subcompartments that meets the preceding definition.

The staff's review includes its evaluation of the distribution of the mass and energy released into the break compartment, the nodalization of subcompartments, the subcompartment vent flow behavior, and the subcompartment design pressure margins. The review of the subcompartment model includes the basis for the nodalization within each subcompartment, the initial thermodynamic conditions within each subcompartment, the nature of each vent flow path considered, and the extent of entrainment assumed in the vent flow mixture. The review may also include an analysis of the dynamic characteristics of components, such as doors, blowout panels, and sand plugs, that must open or be removed to provide a vent flow path, and the methods and results of components tests performed to demonstrate the validity of these analyses. The analytical procedure to determine the loss coefficients and inertia terms [L/A, m^{-1} (ft^{-1})] for each vent flow path, and to predict the vent mass flow rates, is reviewed. The design pressure chosen for each subcompartment is also reviewed.

In Sections 2.6.2 and 2.6.3 of the TPC topical report, DOE has shown that the TPBARs have no significant effect on the mass and energy releases for LOCA or secondary-system pipe ruptures. Therefore, DOE concludes, and the staff agrees, that the containment subcompartment analysis will not be affected by the TPC.

Mass and Energy Release Analysis for Postulated LOCAs (SRP Section 6.2.1.3)

The acceptance criteria for subcompartment analysis are based on the relevant requirements of GDC 50 of Appendix A and Appendix K to 10 CFR Part 50. These requirements include the consideration of the adequacy of the determination of the mass and energy releases to containment from a LOCA, including the following:

(1) energy sources that are available for release to the containment;

(2) mass and energy release rate calculations for the initial blowdown phase of the accident;

(3) for PWRs, because of the additional steam generator stored energy available for release, the mass and energy release rate calculations for the core reflood and post-reflood phases of the accident.

DOE's assessment of the effect of the TPBARs on the LOCA mass and energy releases is summarized in Section 2.6.2 of the TPC topical report. The mass and energy releases in LOCA transients are considered for the short-term subcompartment analysis, and for the long-term containment integrity analysis. 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 is concerned with the performance of the containment heat removal systems and with assuring that the global containment pressure and temperature during a LOCA transient remain within the design limits and equipment qualification profiles.

The evaluation method used by DOE to assess the TPC involved a comparison of the key safety analysis parameters used in a typical Westinghouse LOCA mass and energy analysis for a four-loop plant design for the short-term and the long-term transients. An evaluation of the effect of a full complement of TPBARs indicated a negligible effect on the initial core stored energy. In response to a staff RAI, DOE estimated the increase in the initial core stored energy to be less than 1 percent. The methodology includes a 15-percent margin in the initial core stored energy to determine the mass and energy release from LOCAs. DOE considered the current analysis methodology sufficiently conservative to offset the relatively small increase in the initial core stored energy generated by the TPBARs. In response to a staff RAI, DOE also estimated the increase in the long-term total core heat generation from the TPBARs' gamma heating to be about 0.07 percent. The staff agrees that this can be considered to be insignificant.

A comparison of core and vessel pressure drops with and without the TPBARs was presented in Section 2.3.6 of the TPC topical report. The comparison indicated that the core and vessel pressure drops do change slightly (i.e., less than 3 percent) for normal operation. DOE concluded that the pressure drop changes would not have an effect on the short-term mass and energy release rates and would not be expected to have a noticeable effect on the long-term mass and energy release rates as the system blows down or refills during a LOCA.

An assessment of the minimum containment backpressure calculations for determining the LOCA peak cladding temperature was presented in Section 2.6.4 of the TPC topical report. The installation of the TPBARs and the resulting decrease in primary system liquid volume reduced total mass and energy releases to the containment of approximately 680 kg (1500 lbm—a 0.28 percent reduction) and 8.86×10^5 kJ (839,000 Btu— a 0.27 percent reduction), respectively. These values are a small fraction of the total releases and DOE does not consider them to be significant with respect to the LBLOCA peak cladding temperature or other LOCA limits. In response to a staff RAI, DOE estimated that the reduced RCS inventory (mass and energy released) would result in a reduction in the calculated peak containment pressure of less than 0.69 kPa (0.1 psi). On the basis of a design pressure of 448 kPa (50 psig), this would be equivalent to about a 0.15 percent reduction in the peak calculated pressure. The staff agrees with DOE's assessment that these are not significant with respect to the LBLOCA peak cladding temperature or other LOCA limits.

The incorporation of the TPBARs has been shown to have no significant effect on the mass and energy releases for LOCA pipe ruptures. On the basis of DOE's assessment that none of the key safety analysis parameters are adversely affected for a core reload design with TPBARs, the staff agrees with DOE's conclusion that 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 for the TPC.

However, DOE concludes, and the staff agrees, that a plant-specific evaluation is needed for a TPC if the RCS pressures and temperatures are outside a plant's operating parameters for its current design-basis analyses or if the initial system inventory increased. In addition, a plant-specific evaluation is also needed for a TPC if the current licensing analysis, without the TPC, shows no margin in the calculated maximum containment pressure or temperature.

<u>Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures (SRP Section 6.2.1.4)</u>

The staff's acceptance criteria are based on the relevant requirements of GDC 50 of Appendix A to 10 CFR Part 50. These requirements include consideration of the adequacy of the determination of the mass and energy release to containment from a steamline or feedwater line break, including:

(1) energy sources that are available for release to the containment;

(2) mass and energy release rate calculations;

(3) single-failure analyses performed for the steamline and the feedwater line isolation provisions that would limit the flow of steam or feedwater to the assumed pipe rupture location.

DOE's assessment of the effect of the TPBARs on the secondary-side pipe rupture mass and energy release is summarized in Section 2.6.3 of the TPC topical report.

Mass and energy releases in non-LOCA accidents from 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.

DOE's evaluation method involved comparing key safety analysis parameters used in the standard Westinghouse reload safety evaluation process (WCAP-9273-NP-A). A comparison was made with respect to the high-energy secondary-side line break analyses of record (in the reference plant FSAR). As documented in Section 2.4.3 of the TPC topical report, all of the key safety analysis parameters that make up the reference plant safety analyses (for steamline break and feedline break mass and energy releases) bound the core reload design values with TPBARs. DOE has evaluated the effects on the fuel temperatures associated with the ZIRCALOY™ fuel rod cladding. Small changes in the fuel temperatures have little or no effect on non-LOCA safety analyses when using the Westinghouse LOFTRAN computer program. In particular, calculations for secondary-side line break transients mass and energy releases are dominated by assumptions related to the main steam system design and protection system design and are not sensitive to core-related inputs such as fuel temperatures.

On the basis of DOE's assessment that none of the key safety analysis parameters have been exceeded for the core reload design with TPBARs, DOE concludes that 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 the mass and energy releases and to the associated pressure and/or temperature response analyses, also remain valid.

In response to a staff RAI, DOE could not confirm whether or not other vendors' methods and methodologies are consistent with the current Westinghouse methodologies. Therefore, the staff concludes that other vendor designs require an assessment and, possibly, a reanalysis.

The incorporation of the TPBARs has been shown to have no significant effect on the mass and energy releases for secondary-system pipe ruptures. On the basis of DOE's conclusion that none of the key safety analysis parameters has been exceeded for the core reload design with TPBARs, the staff agrees with DOE's assessment that 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 for the TPC. However, a TPC reload for other vendor designs would require an assessment and, possibly, a reanalysis.

<u>Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies (SRP Section 6.2.1.5)</u>

The staff's acceptance criteria are based on the relevant requirements of 10 CFR 50.46 and Paragraph I.D.2 of Appendix K to 10 CFR Part 50. These requirements include consideration of the conservatism included in the calculation of a minimum containment pressure following a LOCA.

The emergency core cooling system (ECCS) supplies water to the reactor vessel to reflood and cool the reactor core following a LOCA in a PWR. The core flooding rate is regulated by the capability of the ECCS water to displace the steam generated in the reactor vessel during the core reflood period. As the core flooding rate for PWR plants depends directly on the containment pressure, the core flooding rate will increase with increasing containment pressure. As part of the evaluation of ECCS performance, the analysis of the minimum containment pressure that could exist during the period of time until the core is reflooded following a LOCA is reviewed to confirm the validity of the containment pressure used in the ECCS performance capability studies. Assumptions are made regarding the operation of engineered safety features heat removal systems; the effectiveness of structural heat sinks within the containment to remove energy from the containment atmosphere, and other heat removal processes such as steam in the containment mixing with ECCS water spilling from the break in the reactor coolant system; and, in the case of ice condenser containments, the effect of mixing with water from melted ice that drains into the lower containment volume. These assumptions are also reviewed by the staff for all PWR containment types.

DOE's assessment of the effect of the TPBARs on the minimum containment pressure is summarized in Section 2.6.4 of the TPC topical report. In Sections 2.6.2 and 2.6.3 of the TPC topical report, DOE has shown that the TPBARs have no significant effect on the mass and energy releases for LOCA or secondary-system pipe ruptures. Therefore, DOE concludes, and the staff agrees, that the reference plant containment minimum pressure analysis and response will not be affected by the TPC.

Containment Heat Removal Systems (SRP Section 6.2.2)

The staff's acceptance criteria are based on the relevant requirements of GDCs 38, 39, and 40 of Appendix A of 10 CFR Part 50. These requirements include consideration of the adequacy of the systems provided to remove heat from the containment under post-accident conditions. The following specific criteria are reviewed by the staff:

(1) analyses of the consequences of single component malfunctions;

(2) analyses of the available net positive suction head (NPSH) to the containment heat removal system pumps;

(3) analyses of the heat removal capability of the spray water system;

(4) analyses of the heat removal capability of fan cooler heat exchangers;

(5) potential for surface fouling of fan cooler, recirculation, and residual heat removal heat exchangers, and the effect on heat exchanger performance;

(6) design provisions and proposed program for periodic inservice inspection and operability testing of each system or component;

(7) design of sumps and water sources for emergency core cooling and containment spray systems, including an assessment for potential loss of long-term cooling capability due to LOCA-generated debris effects such as debris screen blockage and pump seal failure;

(8) effects of debris, such as thermal insulation, on recirculating fluid systems.

As discussed above, in Sections 2.6.2 and 2.6.3 of the TPC topical report, DOE has shown that the TPBARs have no significant effect on the mass and energy releases for LOCA or secondary system pipe ruptures. Therefore, DOE concludes, and the staff agrees, that the containment heat removal systems will not be affected by the TPC.

Secondary Containment Functional Design (SRP Section 6.2.3)

The staff's acceptance criteria are based on the relevant requirements of Appendix I and GDCs 4, 16, and 43 of Appendix A of 10 CFR Part 50. These requirements include consideration of 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, DOE concludes, and the staff agrees, that there would be no effect as long as the mass and energy releases do not increase as a result of a TPC reload.

Containment Isolation System (SRP Section 6.2.4)

The staff's acceptance criteria are based on the relevant requirements of GDCs 1, 2, 4, 16, 54, 55, 56, and 57 of Appendix A of 10 CFR Part 50, and Appendix K of 10 CFR Part 50. These requirements include consideration of the adequacy of the provisions for containment isolation following a postulated accident.

The staff's review of DOE's TPC topical report regarding containment isolation provisions covered the following aspects:

(1) The design of containment isolation provisions, including:

  (a) number and location of isolation valves, that is, the isolation valve arrangements and the physical location of isolation valves with respect to the containment;

  (b) actuation and control features for isolation valves;

  (c) positions of isolation valves for normal plant operating conditions (including shutdown), postaccident conditions, and in the event of valve operator power; failures

  (d) valve actuation signals;

  (e) basis for selection of closure times of isolation valves;

  (f) mechanical redundancy of isolation devices;

  (g) acceptability of closed piping systems inside the containment as isolation barriers.

(2) protection provided for containment isolation provisions against loss of function from missiles, pipe whip, and earthquakes

(3) environmental conditions inside and outside the containment that were considered in the design of isolation barriers.

(4) design criteria applied to isolation barriers and piping

(5) provisions for detecting a possible need to isolate remotely and manually controlled systems, such as engineered safety features systems

(6) design provisions for and technical specifications pertaining to operability and leakage rate testing of the isolation barriers

(7) calculation of containment atmosphere released before isolation valve closure for lines that provide a direct path to the environs

(8) containment purging/venting design features provided to minimize purging time consistent with principles for maintaining occupational exposure as low as is reasonably achievable (ALARA)

(9) reliability of the purge system to isolate under accident conditions

(10) containment isolation and valve indication provisions in the event of a station blackout (SBO)

These containment isolation system criteria are not affected by the presence of the TPBARs. The selection of the closure times for the isolation valves is based on minimizing the release of the containment atmosphere to the environs, to mitigate the offsite radiological consequences, and to ensure 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. DOE considers tritium to be a minor concern compared to the core activity that is assumed to be released to the containment. Therefore, DOE concludes, and the staff agrees, that the containment isolation systems are not affected by the TPC.

<u>Combustible Gas Control in Containment (SRP Section 6.2.5)</u>

The staff's acceptance criteria are based on the relevant requirements of 10 CFR 50.44, 10 CFR 50.46, and GDCs 5, 41, 42, and 43 of Appendix A of 10 CFR Part 50. These requirements include consideration of the analysis of hydrogen and oxygen production following a LOCA and the capability to monitor, mix, and reduce the combustible gas concentrations in containment.

The major sources of hydrogen and oxygen are

(1) chemical reaction between the fuel rod cladding and steam

(2) corrosion of aluminum and other materials by an alkaline spray solution

(3) radiolytic decomposition of the water in the reactor core and the containment sump

DOE's assessment of the effect of the TPBARs on the combustible gas concentrations, and on the requirements for the combustible gas control systems, is summarized in Section 2.6.5 of the TPC topical report.

As part of the acceptance criteria for the design of the systems provided for combustible gas control, the analyses should indicate that a single-system train is capable of maintaining the combustible gas concentrations to such levels that uncontrolled hydrogen/oxygen recombination would not take place. DOE's assessment of the effect of the TPC on post-LOCA hydrogen generation inside the containment indicated that the TPC will not have a significant effect on the total post-LOCA hydrogen produced in a plant that has operated with a conventional core design.

The TPC can have an effect on the post-LOCA hydrogen generation inside the 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:

(1) metal-water reaction with the fuel cladding;

(2) corrosion of materials;

(3) radiolysis in the sump and core solutions;

(4) RCS inventory prior to the accident.

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

- metal-water reaction with the zirconium components associated with the TPBARs

- tritium and hydrogen that exist in the TPBARs preceding the accident.

Radiolysis, which is a function of decay energy of the fission products, could be marginally affected by the TPC. DOE considered the effect negligible because the TPC has lower burnups than conventional cores (the TPC equilibrium cycle burnup is 25,976 MWD/MTU, as compared to the reference plant Cycle 8 core average burnup of 34,160 MWD/MTU). Corrosion of materials is a function of the post-LOCA temperature inside the containment. In Section 2.6.2 of the TPC topical report, DOE indicates that the associated changes would be negligible, since there was no significant change in the post-LOCA containment temperature with the TPC.

The first potential source of hydrogen, as identified above, that can be attributed to the TPC design is associated with the additional inventory of zirconium in the getter materials contained inside the TPBARs. On the basis of the chemical stoichiometry of the zirconium-water reaction (i.e., 1 pound-mole of zirconium metal reacted must produce 2 pound-moles of hydrogen), 0.224 standard cubic meters (scm) (7.9 standard cubic feet—scf) of hydrogen gas would be produced for each pound of zirconium metal reacted. It is noted, on the basis of the response to a staff RAI, that DOE used 0 °C (32 °F) at 760 mm Hg (1 atmosphere) for the definition of standard temperature and pressure (STP); in the SRP, STP is defined at 21.1 °C (70 °F), leading to 0.240 scm (8.4866 scf) of hydrogen per pound of reacted zirconium. If the total mass of zirconium associated with the TPBARs' getter materials in an equilibrium cycle is converted to hydrogen, the amount of hydrogen gas generated is 215.2 scm (7600 scf).

DOE noted that the zirconium that is subject to the zirconium-water reaction is specified in 10 CFR 50.44 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 the only metal in the active core region that is subjected to high temperatures, in excess of 982.2 °C (1800 °F), where the zirconium-water reaction is expected to occur. However, DOE elected to treat TPBARs in the same manner as active fuel to conservatively assess the combustible gas control systems.

The LBLOCA analysis for the TPC core (Section 2.15.5.2 of the TPC topical report) indicated that a fraction of the TPBARs could exceed 982.2 °C (1800 °F) and would be expected to burst. Following expulsion of the gases, some diffusion of steam into the TPBARs could be postulated. For conservatism, DOE treated the TPBARs' internal zirconium components in a fashion analogous to the treatment of the internal surface of the 3.26 m (128.5 in) fuel rod cladding following cladding burst. For the fuel rod, zirconium oxidation is calculated on the internal surface over the length of the burst node. For the TPBARs, complete oxidation of the zirconium within the 0.076 m (3 in) long burst node was assumed. The equivalent hydrogen that could be released is 5 scm (7600 scf x 3 ÷ 128.5 = 178 scf).

Another potential source of hydrogen and tritium is that contained within the TPBARs. Although considered by DOE to be inconsistent with the criterion as specified in 10 CFR 50.46 (that the calculated fraction of the cladding in the core region subject to the zirconium-water reaction is less than 1 percent), Regulatory Guide 1.7 specifies that a core-melt scenario shall be considered in defining fission-product sources. These sources are used to define the hydrogen generation from the core and the sump solution radiolysis. For conservatism, the core-melt scenario considered by DOE was the release of all the tritium from the TPBARs' getter materials to the containment. The amount of tritium produced in the TPC core designs is 2860 and 2805 grams of tritium for the initial and the equilibrium cycle, respectively (Section 2.4.3 of the TPC topical report). DOE conservatively assumed 0.9 gm/rod (or 3000 gm total for 3344 rods) and, converting this amount of tritium (T) to an equivalent volume of tritium gas (T_2), resulted in a volume of 11.24 scm (397 scf) of T_2. An additional source of hydrogen associated with the TPBARs is the hydrogen generated from the $^{3}He(n,p)T$ reaction inside the rods. At end-of-life, this source could generate an additional 0.255 scm (9 scf) of hydrogen, which would be available for release following a LOCA.

The total amount of the additional tritium and hydrogen associated with a TPC was conservatively estimated by DOE to be 16.54 scm (584 scf). This inventory would be expected to exist in the primary coolant as water or tritiated water (H_2O 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 a gas, the added inventory represents an increase of 8.3 percent in the amount of hydrogen gas that is released from the reactor core 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 percent of the core cladding involved in the reaction (or 15.15 lb-mole) of 154 scm (5440 scf) of hydrogen, and an RCS inventory of 47 scm (1660 scf) of hydrogen before the accident.

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

The lower flammability limit for hydrogen in the containment atmosphere that should not be exceeded (Regulatory Guide 1.7) is 4 volume percent. For the reference plant with a total containment free volume of 7.8×10^{4} m^3 (2,750,000 ft^3), a concentration of 4 volume percent equates to about 2832 scm (100,000 scf) of hydrogen. The contribution of the TPC tritium inventory to the recommended regulatory guide limit is only 0.58 percent (584÷100,000), or less than 1 percent of the total from all sources. For a plant with a much smaller containment volume (for example, an ice condenser plant), the fraction of the total is only about 1 percent (the ice condenser plant containment volume is about half that associated with a plant with a large, dry containment). DOE concluded that even using highly conservative assumptions, the contribution of hydrogen from TPBARs would not exceed 1 percent of the limiting post-LOCA hydrogen inventory inside the containment.

In response to a staff RAI, DOE addressed the margin to the allowable hydrogen concentration. On the basis of the conservatively calculated hydrogen and tritium produced from the TPC of 16.54 scm (584 scf), the hydrogen concentration from the TPC represents about 0.02 weight percent. The range of containment free volume in operating Westinghouse plants is about a factor of 2, with an ice condenser contaiment having about one-half the volume of a large dry containment. Therefore, the worst-case additional hydrogen concentration from the TPC would be about 0.04 weight percent. DOE considered this to be a small value as compared to the 0.5 weight percent margin for the reference plant. The reference plant's FSAR indicates that the actuation of a single hydrogen recombiner at 2 days after a LOCA results in a containment concentration that is below 3.5 weight percent, which is below the 4 weight percent allowable concentration limit – a 0.5 weight percent margin.

In response to another staff RAI, DOE addressed the requirement in 10 CFR 50.44 for assuming the amount of hydrogen produced to be 5 times that produced by the metal-water reaction in 10 CFR 50.46. The increased hydrogen production from the 5 percent metal-water reaction, from 154 scm (5440 scf – at 1.5 percent) to 514 scm (18,140 scf – at 5 percent), is equivalent to a decrease in margin of about 0.5 weight percent. This suggests that the 4 weight percent limit could be reached; however, earlier initiation of the hydrogen recombiners would provide additional margin to the limit and would offset the additional hydrogen production. The magnitude of the sources of hydrogen that are considered in the evaluation of post-LOCA hydrogen generation are plant specific. A plant-specific assessment is required to quantify the sources and to determine the time at which initiation of recombiner operation should commence to limit the hydrogen concentration to acceptable levels.

The staff agrees with the DOE conclusions, based on the conservative assessment of the TPBARs on the combustible gas concentrations in containment following a LOCA, that the combustible gas control systems are not expected to be affected by the TPC. However, the staff concludes that a plant-specific assessment is required to quantify the sources and to determine the time at which initiation of recombiner operation should commence to limit the hydrogen concentration to acceptable levels. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

<u>Containment Leakage Testing (SRP Section 6.2.6)</u>

The staff's acceptance criteria are based on the relevant requirements of GDCs 52, 53, and 54 of Appendix A of 10 CFR Part 50 and Appendix J of 10 CFR Part 50. These requirements include consideration of the preoperational and periodic leak testing of the reactor containment, and of systems and components that penetrate the containment.

The staff's review of the reactor containment leakage testing program covers the following specific areas:

(1) containment integrated leakage rate tests (Type A tests as defined by Appendix J of 10 CFR Part 50), including pretest requirements, general test methods, acceptance criteria for preoperational and periodic leakage rate tests, provisions for additional testing in the event of failure to meet acceptance criteria, and scheduling of tests;

(2) containment penetration leakage rate tests (Type B tests as defined by Appendix J of 10 CFR Part 50), including identification of containment penetrations, general test methods, test pressures, acceptance criteria, and scheduling of tests;

(3) containment isolation valve leakage rate tests (Type C tests as defined by Appendix J of 10 CFR Part 50), including identification of isolation valves, general test methods, test pressures, acceptance criteria, and scheduling of tests;

(4) technical specifications pertaining to containment leakage rate testing reviewed at the operating license (OL) or combined license (COL) stage.

The TPBARs do not affect the containment penetrations or isolation valves. Therefore, DOE concludes, and the staff agrees, that the TPC does not affect the capability to perform the leakage tests as prescribed in Appendix J of 10 CFR Part 50.

2.7 Instrumentation and Controls

Section 2.7 of the TPC topical report addresses various plant instrumentation and control systems. The primary purposes of these systems 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. DOE concludes, and the staff agrees, that the introduction of a full complement of TPBARs into the reference plant will not affect the reactor trip system and engineered safety features actuation system setpoints. In addition, the safe shutdown systems and information systems that are important to safety will not be affected. Therefore, no changes to the instrumentation and controls of the reference PWR are needed to accommodate a TPC.

However, DOE concludes, and the staff agrees, that plant-specific setpoints need to be addressed by licensees participating in DOE's program for the CLWR production of tritium to determine whether or not some of the pertinent parameters might be affected by optimizations performed on a plant-specific basis. In addition, for plants that employ bottom-mounted thermocouples, the process measurement effects previously calculated for post-accident monitoring must be revalidated for the presence of TPBARs.

2.8 Electric Power

Section 2.8 of the TPC topical report addresses the offsite and onsite power systems. The staff's evaluation of the effect of a TPC plant on these sections follows.

SRP Section 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. DOE concludes, and the staff agrees, that the incorporation of TPBARs does not change the grid connections or the applicable acceptance criteria and guidelines.

SRP Section 8.2

Offsite Power System: The acceptance criteria in this section are based on the relevant requirements of GDCs 5, 17, and 18 of Appendix A of 10 CFR Part 50. 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 provision to minimize the possibility 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. DOE concludes, and the staff agrees, that the incorporation of the TPBARs does not affect the capacity or the physical and electrical layout of the offsite system.

SRP Section 8.3.1

AC Power Systems (Onsite): The acceptance criteria in this section are based on the relevant requirements of GDCs 2, 4, 5, 17, 18, and 50 of Appendix A of 10 CFR Part 50. They deal with the adequacy of the onsite ac power system with regard to: (1) the required redundancy, (2) the single failure criterion, (3) protection from the effect of postulated accident, (4) testability, and (5) the capacity and capability to supply power to all safety loads and other required equipment. DOE concludes, and the staff agrees, that the incorporation of the TPBARs does not affect the capacity or the physical and electrical layout of the onsite ac power system. In addition, in its evaluation for SRP Section 3.11 (discussed in Section 2.3.7 of the TPC topical report), DOE determined, and the staff agrees, that the incorporation of the TPBARs will not cause the post-accident environment to exceed the electrical equipment qualification envelope.

SRP Section 8.3.2

DC Power System (Onsite): The acceptance criteria in this section are based on the relevant requirement of GDCs 2, 4, 5, 17, 18, and 50 of Appendix A of 10 CFR Part 50. They deal with the adequacy of the onsite dc 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 dc power to all safety loads and other required equipment. DOE concludes, and the staff agrees, that the incorporation of the TPBARs does not affect the capacity of the physical and electrical layout of the onsite dc power system. In addition, in its evaluation for SRP Section 3.11 (discussed in Section 2.3.7 of the TPC topical report) DOE determined, and the staff agrees, that the incorporation of the TPBARs will not cause the post-accident environment to exceed the electrical equipment qualification envelope.

Conclusion

On this basis, the staff concludes that the incorporation of core reload design with TPBARs will not have any effect on a plant's electrical systems, and it is not anticipated that there would be changes to Chapter 8 of a typical safety analysis report as a result of incorporating TPBARs.

2.9 Auxiliary Systems

Section 2.9 of the TPC topical report addresses the various auxiliary systems required in the plant.

2.9.1 Introduction

The staff has reviewed the evaluations presented by DOE in Section 2.9.1 of the TPC topical report on the effect of the TPBARs on various auxiliary systems against the relevant guidance of the SRP, as described below:

Light Load Handling System (SRP Section 9.1.4)

DOE evaluated the affect of TPBARs on the light load handling system for the reference plant against the guidance of SRP Section 9.1.4. DOE states, and the staff agrees, that the incorporation of the TPBARs has no effect on this system. However, DOE concludes, and the staff agrees, that because of the increase in weight of TPBARs compared to burnable poison rod assemblies, this effect should be evaluated on a plant-specific basis. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

Overhead Heavy Load Handling System (SRP Section 9.1.5)

DOE evaluated the effect of TPBARs on the overhead heavy load handling system for the reference plant against the guidance of SRP Section 9.1.5. DOE states that the incorporation of the TPBARs has no affect on this system. The staff agrees with this evaluation because the spent fuel pool cask bridge crane is the only part of the system that might be used to lift the TPBAR assemblies, and the crane is capable of handling the TPBAR loads.

Station Service Water System (SRP Section 9.2.1)

DOE evaluated the effect of TPBARs on the station service water system (SSWS) for the reference plant against the guidance of SRP Section 9.2.1. The staff's acceptance criteria for the system design are based on meeting the relevant requirements of GDCs 2, 4, 5, 44, 45, and 46 of Appendix A of 10 CFR Part 50. DOE states that the system heat transfer and cooling water flow requirements may be affected by the TPC from the increase of the spent fuel pool heat load during cooldown operations and the subsequent impact on the component cooling water system. However, a quantitative analysis of the effect of the TPC on the SSWS was not performed. In its response to the staff's RAI dated October 15, 1998, DOE also stated that the evaluation of the SSWS heat transfer and flow rate from to TPBARs is extremely plant specific and the extent of the effect depends on available margins in the system. Therefore, a generic evaluation based on the reference plant is not appropriate.

The staff has reviewed the information presented by DOE and concludes that the effect on the SSWS is not safety significant, because the additional heat load introduced by TPBARs is very low and is indirectly transferred to the SSWS. The staff also agrees that, during the generic review of the TPC topical report, a quantitative analysis of the effect of the TPBARs on the SSWS was not appropriate. However, DOE concludes, and the staff agrees, that a quantitative analysis for the SSWS needs to be addressed by licensees participating in DOE's program for the CLWR production of tritium. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

Potable and Sanitary Water Systems (SRP Section 9.2.4)

DOE evaluated the effect of TPBARs on the Potable and sanitary water systems for the reference plant against the guidance of SRP Section 9.2.4 and concludes that there is no effect on this system. The staff agrees with this evaluation.

Ultimate Heat Sink (SRP Section 9.2.5)

DOE evaluated the effect of TPBARs on the ultimate heat sink (UHS) for the reference plant against the guidance of SRP Section 9.2.5. The acceptance criteria specified in the SRP are based on meeting the relevant requirements of GDCs 2, 5, 44, 45, and 46 of Appendix A of 10 CFR Part 50. DOE states that the heat removal capability of the UHS may be affected by the TPC from the increase in the spent fuel pool heat load during cooldown operations and the subsequent effect on the component cooling water system and the station service water system. DOE concludes that the effect on the ultimate heat sink should be analyzed on a plant-specific basis. The staff agrees with this evaluation because the design of the ultimate heat sink is very plant-specific. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

Condensate Storage Facilities (SRP Section 9.2.6)

DOE evaluated the effect of TPBARs on the condensate storage facilities for the reference plant in accordance with the guidance of SRP Section 9.2.6. The acceptance criteria specified in the SRP are based on meeting the relevant requirements of GDCs 2, 5, 44, 45, and 46 of Appendix A of 10 CFR Part 50. DOE states that there is no effect on the condensate storage facilities, so the physical layout, design requirements, and operation of the condensate storage facilities are not required to be modified. The staff agrees with this evaluation.

Compressed Air Systems (SRP Section 9.3.1)

DOE evaluated the effect of TPBARs on the compressed air system for the reference plant against the guidance of SRP Section 9.3.1 and concluded that the incorporation of the TPBARs has no effect on the system operation. The staff agrees with this evaluation.

Equipment and Floor Drainage System (SRP Section 9.3.3)

DOE evaluated the effect of TPBARs on the equipment and floor drainage system for the reference plant against the guidance of SRP Section 9.3.3 and concluded that there is no effect on this system. The staff agrees with this evaluation.

Chemical and Volume Control System (SRP Section 9.3.4)

DOE evaluated the effect of TPBARs on the chemical and volume control system (CVCS) for the reference plant against the guidance of SRP Section 9.3.4. The acceptance criteria for the system design are based on meeting the relevant requirements of GDCs 1, 2, 5, 14, 29, 33, 35, 60, and 61 of Appendix A of 10 CFR Part 50. DOE states that there is no impact on the CVCS function regarding reactor coolant pump seal injection, RCS inventory control, or safety injection or boration functions. The staff concludes that no changes are needed to the CVCS design to accommodate the TPC because there are no changes to RCS parameters, the safety injection function of the CVCS is not affected, the LOCA and other safety analyses are not affected, and the boration functions of the CVCS have been determined to be within CVCS capabilities.

2.9.2 New and Spent Fuel Storage (SRP Sections 9.1.1 and 9.1.2)

The safety function of the storage for new fuel and spent fuel assemblies is to maintain the fuel assemblies in a safe and subcritical array during all credible storage conditions. To ensure adequate safety under normal and accident conditions, the new and spent fuel storage racks are designed for the effects of external loads, forces on the racks, and the effects of natural phenomena such as safe shutdown earthquakes. The reference plant spent fuel pool is safety-related, and is designed with high-density storage racks that are seismic Category I equipment for storing standard fuels (without TPBARs).

The staff reviewed the effect of storing fuel assemblies with TPBAR assemblies in the new and spent fuel racks for the reference plant in accordance with SRP Section 9.1.1 for the new fuel storage and SRP Section 9.1.2 for the spent fuel storage. An analysis has previously been performed using the weight of 1470 pounds for a standard fuel assembly. The TPBARs, as burnable poisons, are similar in form to the Westinghouse standard burnable poison rod assemblies (BPRAs). Because certain space on the storage racks for fuel assemblies will be replaced by TPBAR assemblies, the combined weight of a fuel assembly with TPBARs was calculated to be less than 1430 pounds. DOE also analyzed the dynamic effects for the TPBAR assembly that rests on the top nozzle adapter plate of the fuel assembly and found that the dynamic effect is insignificant. Because the weight of a fuel assembly with TPBARs is less than the weight of the standard fuel assembly previously analyzed, the staff concludes that the current design of the new and spent fuel pool facilities is still valid for the racks containing TPBAR assemblies. However, because the fuel rack analysis is plant-specific, the staff agrees with DOE’s conclusion that the specific storage configuration for a plant participating in DOE’s program for the CLWR production of tritium should be analyzed and could require changes to the TS. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

In addition, DOE did not address the activities required to remove the TPBARs from the fuel assemblies and prepare them for shipment because these activities are dependent on the fuel pool design. Therefore, the staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

2.9.3 Spent Fuel Pool Cooling and Cleanup System (SRP Section 9.1.3)

The spent fuel pool cooling and cleanup system has the safety function to provide adequate cooling to the spent fuel during all operating and accident conditions. The staff reviewed the effect of the TPBARs on the system for the reference plant in accordance with SRP Section 9.1.3. The acceptance criteria for the design of the spent fuel pool cooling and cleanup system and its makeup system are based on meeting the relevant requirements of GDCs 2, 4, 5, 44, 45, 46, 61, and 63 of Appendix A of 10 CFR Part 50.

DOE evaluated the effect of TPBARS on the spent fuel pool cooling and cleanup system for the reference plant against the guidance of SRP Section 9.1.3. DOE states that 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.

The staff agrees with this evaluation.

DOE also performed (as discussed in Section 2.9.3 of the TPC topical report) an analysis of decay heat buildup and the spent fuel pool cooling system (SFPCS) heat removal capability when TPBARs are placed in the spent fuel pool. The operating basis for the reference plant is for 100 percent of the core to be removed during refueling and 60 percent to be reinserted into the core. The design basis of the SFPCS for the reference plant is that one-third of the core is removed on a 12-month refueling cycle. DOE used two approaches to analyze the decay heat load of the spent fuel pool. The first approach estimated the heat load on the basis of two conditions: (1) the actual number of assemblies removed from the core during an18-month fuel cycle; and (2) the removal of the required assemblies during a TPC equilibrium cycle. DOE used the estimated heat loads from the two conditions to calculate the maximum pool temperature and the heat exchanger performance. DOE then compared the estimated maximum pool temperatures for a normal core to that of a TPC. The comparison shows that the maximum pool temperature with a TPC is higher than with a normal core.

The second approach utilized the heat loads from the normal core documented in the reference plant FSAR and compared it with the expected heat loads from the TPC. To assess the effect of TPBARs on the spent fuel pool cooling system, DOE presents a comparison (Table 2.9.3-1 of the TPC topical report) of the maximum spent fuel pool temperature under three conditions: (1) normal refueling vs. normal with TPBARs refueling, (2) maximum normal refueling vs. maximum normal with TPBARs refueling, and (3) maximum normal refueling under emergency conditions vs. maximum normal refueling with TPBARs under emergency conditions. The greatest effect of the TPC was found during normal refueling because a significant portion of the TPC (about 75 percent, as stated in Section 2.9.3.1 of the TPC topical report) will be removed and the heat load increase is the highest. The maximum spent fuel pool temperature was 139.3 °F for the refueling with normal core and 160 °F for normal refueling with a TPC. An increase in the maximum pool temperature by up to 21 °F was found for the normal refueling with TPBARs. In response to a staff RAI, DOE stated that the temperature increase in the spent fuel pool comes from the insertion of additional fuel assemblies removed from the core every cycle and the heat load from an entire core with or without TPBARs is the same. A detailed analysis of the capability of the spent fuel pool cooling system to maintain the bulk average temperature was not performed because of the large variation in the capacity of the spent fuel pool and its associated cooling system design between plants.

The staff has reviewed the information presented by DOE and concludes that the calculations performed by DOE may not represent the actual increase in pool temperature from incorporation of the TPBARs. However, on the basis of information submitted by DOE in its letter dated January 13, 1999, the decay heat generated by the TPBARs is very low; each TPBAR generates less than 3 watts of heat at 150 hours after reactor shutdown. The maximum temperature increase of a TPBAR due to internal heat generation is less than 3 °F. The reference plant could insert up to 3344 TPBARs in each reload. The total heat load increase due to TPBARs is about 0.003 percent compared with a 3565 MWT core rating of the reference plant. In considering its very low rate of heat generation, the staff concludes that the heat load increase from the incorporation of TPBARs in the spent fuel pool has an insignificant impact on the spent fuel pool heat load and the added heat load will be within the cooling capability of the SFPCS. However, further analysis with reliable data is required to determine the actual impact of the TPBARs. A quantitative analysis to determine the absolute spent fuel pool temperatures must be performed by licensees seeking to utilize a TPC because the capacity of the spent fuel pool and its associated cooling system design are very plant specific. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

2.9.4 Component Cooling Water System (SRP Section 9.2.2)

The component cooling water system (CCWS) of the reference plant supplies cooling water to the residual heat removal (RHR) heat exchangers, the motor coolers of the RHR pumps, and the spent fuel pool heat exchangers. The CCWS heat exchangers are cooled by the nuclear service cooling water system. DOE evaluated the effect of the TPBARs on the CCWS against the guidance of SRP Section 9.2.2. The acceptance criteria for the system design are based on meeting the relevant guidance of GDCs 2, 4, 5, 44, 45, and 46 of Appendix A of 10 CFR Part 50.

Because more fuel and TPBAR assemblies are removed from the core to the spent fuel pool during refueling, the maximum pool temperature will increase. Although the effect of the TPBARs on the CCWS is insignificant because the heat load generated by the TPBARs only amounts to about 3 watts per rod 150 hours after reactor shutdown, a substantial increase in heat load occurs as a result of a full core off-load. The additional heat load generated by the TPC to the spent fuel pool heat exchangers could increase the demand for CCWS flow. DOE stated that the system heat transfer and flow requirements may be affected by the TPBARs from the increase in spent fuel pool heat load during cooldown operations, and the effect on this system will need to be analyzed on a plant-specific basis. In response to the staff's RAI, DOE also stated that the increased spent fuel pool heat load does not come from the presence of TPBARs but from the increased number of fuel assemblies being replaced. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

2.9.5 Demineralized Water Makeup System (SRP Section 9.2.3)

The demineralized water makeup system of the reference plant is non-safety-related and is designed to provide an adequate supply of treated water of reactor coolant purity to other systems as makeup. DOE evaluated the effect of TPBARs on the demineralized water makeup system against the guidance of SRP Section 9.2.3. With the incorporation of TPBARs, the tritium level in the reactor coolant system (RCS) will increase and the water additions to the RCS for boron concentration control will also account for diluting the tritium. In order to maintain the desired tritium concentration, more reactor coolant makeup or more demineralized water flow is required. As a result, the need for makeup from the system will increase. In response to the staff's RAI, DOE stated that, based on the expected increase in total volume required, the demineralized water makeup system will be operated more frequently, but the equipment size and the flow rates will not be affected.

The staff has reviewed the information presented by DOE and concludes that the incorporation of TPBARs in the reference plant does not have any significant impact on the demineralized water makeup system because only a very small quantity of tritium is released from the TPBARs to the primary coolant system. Because the design of the demineralized water makeup system is plant-specific, DOE concludes, and the staff agrees, that a detailed analysis for this effect is required from licensees participating in DOE's program for the CLWR production of tritium. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

2.9.6 Process Sampling System (SRP Section 9.3.2)

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.

The staff reviewed the TPBAR sampling requirements that exceed the normal plant sampling requirements. The staff concludes that there will be a need for increased focus on tritium and lithium analyses. The samples are expected to be obtained from the reference plant's existing RCS liquid sampling lines in the sample room.

Because the TPBARs are introduced to the spent fuel pool, the potential exists for leaching of chemical contaminants, such as aluminum. Spent fuel pool contents are sampled locally on a periodic basis to verify that the minimum boron concentration is maintained, and that the maximum levels of contaminants are not exceeded. The current sampling frequency should be increased during, and immediately after, refueling with TPBARs to ensure compliance with existing spent fuel pool specifications. It is possible that analyses for all chemicals, including tritium, could be performed with the same sample line.

The staff agrees with the TPC topical report that no additional sample points will be needed, and therefore, there is no effect on the sampling system design.

2.9.7 Conclusion

The staff has reviewed the effect of the TPBARs on the preceding auxiliary systems for the reference plant in accordance with the relevant SRP sections. Because the heat load generated by the TPBARs during cooldown operations is very low and the TPBAR assembly is similar in form and weight to those of the standard Westinghouse burnable poison rod assembly, the staff concludes that the effect of the TPBARs on the auxiliary systems is not safety-significant and will be bounded by current design parameters of the reference plant. The staff also finds that the evaluation of the effect of the TPBARs is extremely plant specific, and the extent of the effect depends on the system design parameters. Therefore, as discussed throughout this section, the staff concludes that a quantitative analysis with actual design margins is required for the auxiliary systems by licensees seeking to utilize a TPC.

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. On the basis of DOE's qualitative evaluations, the staff concludes that further evaluations are not required and that it is unlikely that changes to Chapter 10 of a typical SAR would be needed to accommodate a tritium production core.

2.11 Radioactive Waste Management

This section of the SER documents the NRC staff's review of Section 2.11 of the TPC topical report, which addresses the subjects of source terms, liquid waste management systems, gaseous waste management systems, solid waste management systems, and process and effluent radiological monitoring and sampling systems evaluation. The staff reviewed Section 2.11 of the TPC topical report and DOE's responses to the staff's RAIs against the criteria of Sections 11.1 through 11.5 of the SRP.

2.11.1 Source Terms

Section 2.11.2 of the TPC topical report describes the effect of the TPBARs on the source terms. Section 11.1 of the SRP contains the acceptance criteria for the staff's evaluation.

Section 2.11.2.2 of the TPC topical report states that the higher enrichments and reduced fuel assembly burnups associated with TPC core design, as compared to conventional core design, can affect the design-basis source terms. To assess the effect, DOE used the ORIGEN2 computer code to calculate the TPC fission-product nuclide inventories in the reactor core and the major fission-product activity concentrations in the primary coolant. These include noble gas and iodine isotopes, long-lived species such as Kr-85, Cs-134, Cs-137, and Sr-90, and other isotopes identified in Tables 2.11.2-1 and 2.11.2-2 in the TPC topical report and Tables 2.11.2-1a and 2.11.2-2a in the response to RAI G.14. These were compared to similar calculations made for the reference plant operating with a conventional core. On the basis of the comparisons, DOE concludes that, for the design-basis source terms, 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 comply with the applicable regulatory requirements of 10 CFR Part 20, Appendix I of 10 CFR Part 50, and GDC 60 of Appendix A of 10 CFR Part 50.

The operation of a TPC can increase the amount of tritium released into the primary coolant. DOE analyzed the projected increases in plant tritium activity production and effluents. 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. DOE established a design goal of less than 1.0 Ci/yr/TPBAR released to the primary coolant. There will be monitoring and surveillance programs to ensure the achievement of the design goal. On the basis of the design goal of 1.0 Ci/rod/yr from TPBARs for a plant operating a full complement of TPBARs, the potential increase of tritium inventory in the primary coolant could be from a nominal value of 890 Ci/yr to 4268 Ci/yr for a typical four-loop PWR with a power level of 3565 MWt. The design-basis tritium sources are expected to increase the amount of tritium that is discharged annually by a factor of about 5.

In addition to the design-basis releases from the TPBARs by permeation processes, another potential release scenario is the failure of the TPBARs, so that the inventory in the TPBAR is released into the primary coolant. The assumed failure rate for this release scenario is the simultaneous failure of two rods, which is equivalent to a total of 20,000 Ci. In response to a staff RAI, DOE stated that the assumption was based on its operating experience data from other types of burnable poison rods and that the data are applicable for TPBARs because a similar quality standard is applied in the fabrication of those rods.

On this basis, the staff concludes that operations with TPBARs in the core does affect the radioactive source terms because it results in an increase in the amount of tritium in the reactor coolant system (RCS) and within the plant. Therefore, a TPC causes an increase in plant liquid and gaseous tritium effluent releases and solid radwaste. The impact of these increases on plant systems from both design-basis tritium by permeation and tritium released from TPBAR failures is evaluated in Sections 2.11.2 – 2.11.5 of this report.

2.11.2 Liquid Waste Management Systems

Section 2.11.3 of the TPC topical report describes the impact of tritium increases in the primary coolant of a TPC plant on the liquid waste management systems. Section 11.2 of the SRP contains the acceptance criteria for the staff's evaluation.

The tritium in the primary coolant exists as tritiated water, which cannot be removed by normal waste processing techniques such as ion exchange, filtration, or evaporation. Because of 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. 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 plants through the release of tritiated water is the most practical means available. To address tritium buildup concerns with a TPC, much larger volume of primary coolant would have to be discharged as compared to current operating plant practices. These releases are batch releases via a monitored tank that is sampled before the 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 based on the activity concentrations in the tank and the flow rate of the dilution water, and is controlled so that the diluted concentration does not exceed applicable regulatory and plant technical specification limits. The reference plant technical specifications refer to the plant's Offsite Dose Calculation Manual (ODCM), and the ODCM reflects the applicable criteria of 10 CFR Part 20 and Appendix I of 10 CFR Part 50.

The effluent concentration limit in water from Appendix B and Table 2, Column 2 of 10 CFR Part 20 is $1x10^{-3}$ μCi/ml. However, the reference plant ODCM indicates that a release of 10 times this 10 CFR Part 20 concentration would be permitted based on an exemption granted by the NRC. DOE indicates that similar exemptions have been granted to other PWR plants. The staff agrees with DOE's approach to evaluate the reference plant with respect to this criterion. However, the staff concludes that the applicable plant-specific values of effluent concentration and dose limits as well as dilution flow rate need to be evaluated by licensees participating in DOE's program for the CLWR production of tritium. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

<u>Design-Basis Tritium</u>

On the basis of the design-basis tritium sources of 4268 Ci/yr in the primary coolant, DOE determined the amount of liquid discharge that would be required to maintain reasonable tritium concentration levels in the reference plant by using a computer model to simulate tritium production, release, and mixing in various water volumes. DOE found that the reactor coolant concentrations in the range of 1.5 – 2.0 μCi/gm will be maintained if the amount of RCS liquid release is increased to 10 system volumes/yr. On the basis of the 15,500 gpm dilution flows 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 calculated to be $1.5x10^{-4}$ μCi/gm. This concentration is a factor of 6.7 less than the 10 CFR Part 20 limit and a factor of 67 less than the reference plant technical specification limit.

DOE assessed the potential impact of TPC operation on the dose to the public. The relative contribution to the total dose from tritium releases was considered. The projected dose, in terms of the ODCM limits, shows that the fraction of the ODCM limit increases from approximately 1 percent with conventional core designs to 2 percent with TPC operations. Because the primary coolant tritium levels are not changing (since additional batch releases will be made), the doses in airborne effluents are not expected to be affected, and would remain below (e.g., less than 0.005 percent) the ODCM dose limit. In addition, an increased discharge rate of primary coolant will result in a shorter residence/decay time for nuclides in the liquid waste processing systems. As a result, an increase in the activity associated with short-lived isotopes can be anticipated. The increase of activity in liquid effluents was calculated to be less than 5 percent in total curies released, which was found to be insignificant. With this increase, the estimated annual release is still a small fraction of the 10 CFR Part 50, Appendix I limit on liquid pathway releases.

Because the discharges of liquid effluents are made in a batch mode, the increased amount of primary coolant that must be processed and discharged will affect plant waste management and operations. The projected batch discharge frequency with a TPC would increase from about 60 releases to 105 batch releases. In terms of time of releases, this converts to an increase from 130 h/yr to 306 h/yr.

<u>TPBAR Failures</u>

DOE also analyzed the effect of the tritium release from the failure of two TPBARs, so that the inventory of two TPBARs is released to the primary coolant. The inventory in two failed TPBARs was determined to be 20,000 Ci per cycle. The projected release, in terms of the ODCM limits, is approximately 10 percent of the applicable limit for liquid effluents and 0.1 percent of the airborne limit. The staff finds these resultant releases acceptable for the reference plant. However, when the TPC topical report is applied to a candidate plant, a plant-specific evaluation is needed because plant-specific ODCM limits are used for the basis of acceptance.

The RCS tritium concentration could potentially increase rapidly to 89 μCi/g. An increased RCS tritium concentration of this magnitude would not necessarily preclude continued plant operation. However, it may severely limit or restrict access inside the containment until the tritium concentration is reduced to an acceptable level. To reduce the RCS tritium concentration to no more than the maximum recommended concentration of 3.5 μCi/g, feed-and-bleed processing has to be increased or temporary storage has to be added with subsequent processing of primary coolant. The time required to reduce the concentration to 3.5 μCi/g at a letdown flow of 120 gpm is about 27 hours. During the release, the maximum dilution flow rate and/or reduced discharge flow rate should be used so that the discharge concentration limits are met. To properly track and manage any unanticipated events, such as TPBAR failure, DOE found that it would be necessary to develop appropriate procedures and action plans to trigger the increased data monitoring, to initiate recovery actions, and to minimize the impact on doses to workers and members of the public. In response to a staff RAI, DOE discussed the different operator actions corresponding to four levels of tritium concentration. The response actions will be plant-specific and defined in action plans unique to the organization and management structure of the plant.

On the basis of the preceding discussion, the staff concludes that in both cases (the design-basis TPBAR permeation of tritium and the failure of two TPBARs) there is a sufficient margin in the reference plant so that the applicable release concentration and dose limits as presented in the plant technical specifications and ODCM will still be met even with the TPC operation. However, enhanced plant-specific tritium monitoring and surveillance programs and procedures for operator actions on an abnormal tritium release event are required. Furthermore, when the TPC topical report is applied to a candidate plant, a plant-specific analysis will be needed to demonstrate that the plant continuously meets release concentration and dose limits. The staff concludes that the methodology described in Section 2.11.3 of the TPC topical report is acceptable for the plant-specific analysis. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

2.11.3 Gaseous Waste Management Systems

The staff reviewed Section 2.11.4 of the TPC topical report, wherein DOE states that the control strategy for the buildup of tritium does not consider the use of waste gas systems. Therefore, there should be no change between operations with TPC versus a conventional core. 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 μCi/g to the maximum recommended concentration of 3.5 μCi/g, the doses from airborne tritium releases remain a negligible fraction of the ODCM limit from less than 0.005 percent to less than 0.1 percent with TPC operation. Normal evaporative loses from the refueling cavity water and the spent fuel pit are the release pathways. The staff agrees with DOE's assessment that the amount of increase in the radioactive gaseous effluents and dose limits are insignificant, provided that the control strategy for the buildup of tritium does not include the use of waste gas systems.

2.11.4 Solid Waste Management Systems

Section 2.11.5 of the TPC topical report describes the effect of tritium increases in the primary coolant of a TPC plant on the solid waste management systems. Section 11.4 of the SRP contains the acceptance criteria for the staff's evaluation.

The additional liquid releases to control the tritium buildup within the plant may require additional ion exchange and filtration to reduce the radioactivity and contaminant concentrations before discharge, thereby increasing the amount of solid radwaste such as spent resins. DOE estimated that the additional number of resin bed changes due to the design-basis tritium source by permeation from the use of TPBARs is approximately 1/yr. In response to the staff's RAI dated October 15, 1998, DOE stated that for the bed volume of 30 ft^3, it represents a 0.15 percent increase over the estimated annual quantity of total solid radwaste of 20,000 ft^3 for the reference plant. Consideration of two failed TPBARs results in an increased number of resin bed changes from 1 to approximately 4 per year, which still represents an insignificant increase in the total amount of radwaste. The estimated low-level solid waste activity increase is from 2000 Ci to 2600 Ci. The resins are packaged and shipped for ultimate disposal. There is no associated increase in exposure to the public. Further, the resins are processed and packaged so that occupational radiation exposure levels are maintained at levels consistent with as the ALARA principle. Therefore, DOE concludes in its response to the staff's RAI dated October 15, 1998, that the additional activity inventory in the solid radwaste will not compromise the reference plant's compliance with applicable regulations. On the basis of its review of the preceding information, the staff agrees with DOE's conclusion.

2.11.5 Process and Effluent Radiological Monitoring and Sampling Systems

In Section 2.11.6 of the TPC topical report, DOE states that 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, and are adequate for use when the plant is operated with a TPC. On the basis of its review, the staff agrees with DOE that the existing capability for radiation monitoring is adequate for tritium levels at the reference plant. In response to the staff's RAI dated October 15, 1998, DOE stated that the details of the laboratory instrumentation and sampling frequencies and locations are plant dependent. Therefore, a plant-specific assessment of the candidate plant for the TPC will be required to provide such information. The staff has identified this as an interface item that must be addressed by a licesee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

2.11.6 Summary

On the basis of its review of Section 2.11 of the TPC topical report, the staff concludes that

- The methodology described in Section 2.11 of the TPC topical report is acceptable for evaluating the impact of the plant operation with a TPC on the radioactive waste management.

- The major impact on the source terms that could result from the operations with a TPC is an increase in the amount of tritium in the RCS, which would result in an increase in plant liquid and gaseous tritium effluent releases and solid radwaste. The increase in the radioactive gaseous effluent and solid radwaste is insignificant at the reference plant.

- The increased amount of primary coolant that must be processed and discharged will affect plant liquid waste management and operations. There is a sufficient margin in the reference plant so that the applicable release concentration and dose limits as provided in the plant technical specifications and ODCM will be met, even with the increases of radioactive liquid effluents resulting from the operation with a TPC. However, as discussed in Section 2.11.2 if this report, enhanced plant-specific tritium monitoring, and surveillance programs and procedures for operator actions on abnormal tritium release events, are required to be submitted by licensees participating in DOE’s program for the CLWR production of tritium.

- The current process and effluent radiological monitoring instrumentation and sampling systems that are in place at the reference plant include the capability for monitoring the tritium levels within the plant and in plant effluent pathways, and are adequate for use when the plant is operated with a TPC.

- Although the impact on the radioactive waste management systems resulting from the reference plant operating with a TPC is acceptable, a plant-specific evaluation of the candidate plant operation with a TPC will be required as described in Sections 2.11.2 and 2.11.5 of this report.

2.12 Radiation Protection

2.12.1 Introduction

Section 2.12 of the TPC topical report describes the plant programs and design features that are intended to maintain radiation exposures, to plant workers working within the plant and to members of the public exposed to radioactive effluents released from the plant, to levels that are as low as is reasonably achievable (ALARA). Sections 12.1, 12.2, 12.3, 12.4, and 12.5 of the SRP contains the acceptance criteria for the staff’s evaluation.

2.12.2 Radiation Sources

Section 2.12.2 of the TPC topical report describes the effect of operation with TPBARs on radiation sources in normal operations, anticipated operational occurrences, and accident conditions. DOE states that the operation of a plant with a TPC is expected to have negligible effect on the design basis and realistic fission and corrosion product sources and the treatment of these isotopes in gaseous and liquid wastes. In addition, the TPC is not expected to affect the fission-product source terms that are used for shield design, equipment qualification, system design, and accident dose analysis.

The design-basis tritium sources are expected to increase the amount of tritium that is discharged annually by a factor of about 5, that is, from about 890 Ci/yr to the TPC design-basis value of approximately 4300 Ci/yr. However, as discussed in Section 2.11.2 of the TPC topical report, this additional tritium inventory will not interfere with the ability of the plant to meet the requirements of 10 CFR Part 20, GDC 60 of Appendix A of 10 CFR Part 50, and the plant technical specifications to maintain radioactive effluents ALARA. Even with the postulated failure of two TPBARs, in which an additional 20,000 Ci of tritium would be released into the reactor coolant system, the plant's existing waste management equipment is expected to be able to effectively control the release of the tritium into plant effluents without exceeding regulatory requirements.

On this basis, the staff finds that although operation with TPBARs in the core of the reference plant does increase the radioactive source term, existing radioactive waste treatment systems are capable of ensuring that there is negligible impact on the ability of the plant to operate within regulatory requirements.

2.12.3 Radiation Protection Design Features and Dose Assessment

Section 2.12.3 of the TPC topical report describes the impact of plant operation with TPBARs on radiation protection design features, taking into account design dose rates, anticipated operational occurrences, and accident conditions.

The increased inventory of tritium released from the reactor coolant system from the TPBARs and the additional IFBA fuel rods were evaluated to determine whether normal releases can be maintained within regulatory limits as well as being ALARA to plant workers. These criteria are related in that if all of the tritium produced is retained in the plant rather than released in routine plant effluents, the resulting tritium inventory can result in increased radiation dose to plant workers. Important plant activities that can be affected include containment access during power operation and refueling operations.

During power operation, any leakage from the primary system into the containment building could result in higher concentrations of tritium. Also, during refueling operations, the tritium contained in the RCS is dispersed into the water in the refueling cavity, the fuel transfer canal, and the spent fuel pool. The tritiated water that evaporates into the air at these locations can expose plant workers to a radiation dose from inhalation and absorption through the skin.

The preceding scenarios were evaluated in the TPC topical report for their effect on worker radiological exposure. The DOE analysis concluded that there will be a negligible increase in the annual radiological dose to a plant worker from operation with TPBARs. This is based on the recommended procedure to adjust the plant effluent discharge of the tritiated primary coolant so that buildups 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 the use of plastic protective clothing or a self-contained breathing apparatus for protection against airborne tritiated water vapor). The adjustment in plant effluent discharge will depend, in part, on the actual permeation of tritium from the TPBARs, the amount of normal systems leakage at any given point in the plant, and normal waste discharge practices for a particular plant.

The preceding evaluation of the effect 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 percent 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 effluent discharges maintains primary coolant tritium concentrations at typical current levels and does not result in significant increases in offsite doses to members of the public (i.e., generally less than 2 percent 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 fuel and TPBAR handling activities. 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 the associated occupation radiation exposure 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 occupational radiation exposure are about the same, regardless of whether or not a TPC is used. The differences in the occupational radiation exposure for the remaining handling operations, that is, with TPBARs versus a normal core refueling, are highly plant dependent. 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. DOE estimates that the total occupational radiation exposure projected for the additional TPBAR handling is 110 millirem per fuel cycle, which equates to roughly 0.073 person-rem/yr (for an 18-month fuel cycle).

On this basis, the staff agrees with DOE’s assessment that operation with TPBARs in the core will result in a negligible increase in the annual radiological exposure to plant workers. The only potential source of additional exposure associated with TPC operation is that associated with worker radiological exposure from increased fuel and TPBAR handling activities. The total projected dose of 0.073 person-rem/yr is not significant when added to the industry norm of approximately 150 person-rem/yr for routine plant operation.

2.12.4 Operational Radiation Protection Program Evaluation

Section 2.12.4 of the TPC topical report describes the operational aspects of the radiation protection program (organization, equipment, instrumentation, facilities, and procedures) needed to support plant operation with TPBARs.

The reference plant, as with all operating PWRs, currently has an operational health physics organization along with appropriate equipment, instrumentation, facilities, and procedures in place for monitoring of tritium both within the plant and in plant effluents. Plant operation with the TPC can increase the amount of tritium within the plant and in plant releases. However, current health physics programs are expected to be able to handle the potential increase in tritium levels.

On this basis, the staff agrees with DOE’s assessment that operation with TPBARs in the core can be adequately managed by current health physics programs.

2.13 Conduct of Operations

The TPBARs have been designed to be transparent to plant operations. After they are loaded into the fuel assemblies in the proper locations in the reactor, they are an integral part of the core, they are addressed in the core operating limit report, and operation with a TPC core is appropriately limited by the plant technical specifications.

2.13.1 Introduction

Section 2.13.1 of the TPC topical report addresses the effect of operating a reactor facility with a TPC on various aspects of plant operations, including organization, training, emergency planning, operation review, procedures, or security. The staff concludes that these aspects of plant operation are not significantly affected by the use of the TPBARs. The staff's evaluation of security considerations for a reactor facility with a TPC follows.

2.13.2 Safeguards and Security (SRP 13.6)

In Section 2.13.2 of the TPC topical report, DOE addresses transportation and physical security aspects of the TPBAR lead test assemblies (LTAs). DOE states that the TPBARs and some related documentation necessary for utility nuclear safety committee review will be classified "confidential restricted data." As classified matter, they require measures to prevent diversion of, unauthorized access to, and disclosure of classified information.

Materials Control and Accountability

DOE requires that the TPBARs be controlled and accounted for because of the initial presence of lithium-6 and, post-irradiation, tritium. 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. The host utility's internal control and accountability procedures for fuel should be adequate and consistent with the material control and accountability for the TPBARs and assemblies.

Physical Security of Classified Hardware

The TPBARs require physical protection commensurate with their classification as confidential restricted data (CRD). As one of the requirements for an operating license, the host utility will have performed a vulnerability assessment for physical protection of vital equipment. The design-basis threats of the DOE for CRD and the NRC for fuel are similar, and little or no modification of the utility's physical security is expected.

The TPBARs will be brought to the site by a DOE-approved carrier that meets Department of Transportation requirements for shipment of nuclear fuel. Once inside the protected area, personnel who have DOE clearances will monitor the movement of the TPBARs. While the TPBAR LTAs are stored in the new fuel storage racks or in the fuel pool, a suitable level of protection will be provided. When 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.

Control of Classified Documents and Hardware

Because the TPBAR hardware and certain of the documentation are classified, licensees undertaking irradiation of TPBARs will have to meet the requirements for access to CRD that are specified in 10 CFR Parts 25 and 95, and 10 CFR 50.37. By letter dated October 4, 1996, DOE advised the staff that a limited number of licensee employees at Westinghouse Nuclear Fuels and at the Tennessee Valley Authority (TVA) needed access authorization in order for them to perform their responsibilities in support of the DOE CLWR Tritium Project. DOE proposed that it perform the necessary personnel security clearance function and process a limited number of "L" and "Q" access authorizations for these licensee employees. By letter dated November 1, 1996, the staff agreed with DOE's proposal and stated that no additional NRC clearance is required to satisfy the requirements of 10 CFR 50.37 and 54.17(g). This is consistent with the memorandum of understanding between the NRC and DOE, dated September 19, 1996, concerning provisions of the National Industrial Security Program.

DOE has stated that no classified documents related to the TPBARs will be maintained on site at Watts Bar or at TVA headquarters. A reading room is being maintained at Oak Ridge National Laboratory so that individuals with a "need to know" will have access to the classified documents associated with the CLWR Tritium Project.

With regard to the facility (security) clearance (FCL), following discussions between the DOE Office of Safeguards and Security and the NRC Division of Facilities and Security, the staff and DOE have agreed to allow DOE to perform the "cognizant security agency" responsibilities applicable to the protection of classified matter at NRC-licensed facilities involved with the TPBAR LTA irradiation. These facilities include the Westinghouse fuels facility at Columbia, South Carolina, and TVA's Watts Bar plant. (The functions of the "cognizant security agency" are delineated in the "Memorandum of Understanding Between the Department of Energy and the United States Nuclear Regulatory Commission Under the Provisions of the National Industrial Security Program," dated September 19, 1996.) In its letter dated April 21, 1997, the staff summarized the agreement and stated that DOE would have authority over the FCL at the Westinghouse-Columbia and Watts Bar facilities during the LTA irradiation phase of DOE's program for the production of tritium in CLWRs. As agreed, DOE will provide the NRC copies of the DOE-approved security plans for these facilities, invite NRC to participate in facility security reviews, and keep the NRC Division of Facilities and Security fully and currently apprised of security and classification matters at these facilities. The letter also informed DOE that the agreement and the decision regarding future overall security responsibility for this program would be re-evaluated after the LTA irradiation phase was completed.

Following additional discussions with DOE, the staff, in a letter dated March 5, 1999, notified DOE that it had determined that, based on the preponderance of responsibility for the classified aspects (e.g., personnel clearances, responsibility for control of classified information, transportation, and the oversight process), DOE should continue to have the cognizant security agency responsibilities for the FCL activities necessitated by the presence of TPBARs during the production phase of DOE's tritium program at TVA's Watts Bar and Sequoyah facilities. NRC's security oversight and responsibilities will remain the same as at all other CLWRs.

2.14 Initial Test Program

The initial test plan is designed to demonstrate that components and systems operate in accordance with design requirements. 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 or its support systems for handling and processing waste effluents. DOE concludes, and the staff agrees, that operation of the plant with the TPC is not significantly different from the operation of a non-tritium producing core.

2.15 Accident Analysis

Section 2.15 of the TPC topical report addresses the analyses of the anticipated operational occurrences and postulated accidents addressed in Chapter 15 of the SRP. 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.

2.15.1 Introduction

The sections in Chapter 15 of the SRP are related to the analyses of a specific set of anticipated operational sequences and postulated accidents. These analyses include not only the transient analyses, but also the radiological consequences of those accidents that could result in the release of radioactive materials. In Section 2.15 of the TPC topical report, DOE has considered the effect of a TPC on transient and accident analyses, and their radiological consequences.

2.15.2 Non-LOCA Accidents

The effect of the TPBARs on the FSAR Chapter 15 non-LOCA accident analyses for the reference plant has been reviewed by DOE to determine which events need to be reanalyzed. The review was based on event-specific sensitivities and a decision was made for each transient with regard to the need for a formal analysis as opposed to simply evaluating the impact of the subject features and assumptions. As discussed in Section 2.4.3 of this report, even for failures involving a breach of the cladding in two failed TPBARs, the TPBARs would continue to provide sufficient reactivity control. As discussed in Section 3.5 of this report, breach of the TPBAR cladding during normal operation and AOOs is unlikely. Therefore, the failure of two rods at the same time is considered to be a highly unlikely abnormal event. This is supported by the extremely low number of observed failures in commercial burnable absorbers (DOE letter dated October 26, 1998). The only design-basis event that could credibly result in multiple, widespread TPBAR failures is the LBLOCA discussed below.

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 transient, 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 be a valid licensing basis for the plant. When any safety-related parameter is not bounded, further evaluation or reanalysis is needed.

The basic methodology and computer codes used for the analyses are the same as documented in the reference plant FSAR. These include LOFTRAN (WCAP-7907-P-A), FACTRAN (WCAP-7908-A), and TWINKLE (WCAP-7979-P-A).

The nominal plant operating conditions (power, coolant temperature, pressure, and flow rate) are unaffected by the inclusion of the TPBARs with the exception of core bypass flow, which decreases to about 6 percent as compared to the reference plant value of 8.4 percent. However, the reference plant value of 8.4 percent, which assumes that thimble plugs are removed, is retained for the TPBAR evaluations. The staff considers this to be acceptable because it conservatively results in higher core average and core outlet temperatures. The evaluations also assumed 10 percent uniform steam generator tube plugging.

No changes to the thermal hydraulic characteristics or power peaking factors which could affect the core thermal limits have been identified as a result of the use of TPBARs. Therefore, the plant thermal limit protection system setpoints remain the same.

As a result of the nuclear design and fuel rod design calculations performed for the TPBAR reload core design, the BOL Doppler coefficient is slightly less negative than for the reference core. The reduced-thickness IFBA coating results in a slightly larger pellet-to-cladding gap and thus an increase in the maximum fuel pellet average and surface temperatures. Therefore, the accidents sensitive to these parameters have been reevaluated or reanalyzed.

DOE has evaluated each event category of the non-LOCA transients and accidents documented in the reference plant FSAR with regard to the effect of the TPBAR core design using the methods discussed above. For most event scenarios, DOE concluded that the TPBAR design has not changed any of the bounding values assumed for the key safety analysis parameters used in the reference plant FSAR analyses. Also, on the basis of existing sensitivity studies performed for a representative Westinghouse plant, DOE determined that the slightly increased maximum fuel temperatures considered for the TPBAR core do not significantly affect the results for these events. Therefore, all safety analysis acceptance criteria for those event scenarios are met by the TPBAR core design. However, DOE has identified several transients and accidents in which there is either a change of bounding values assumed for the key safety analysis parameters or that are affected by the increase in fuel temperatures. These events are reanalyzed to confirm that the acceptance criteria for each event are still met.

The staff has reviewed each of the accidents that were reanalyzed or reevaluated. These reanalyses applied methods that have been previously found acceptable by the NRC. The results show changes in the consequences of transients and accidents previously analyzed. However, the results remain within the required acceptance criteria. Specifically, for non-LOCA events, during normal operation and anticipated operational occurrences, there is at least a 95 percent probability at a 95-percent confidence level (95/95 probability/confidence) that DNB will not occur on the limiting fuel rod. During these operational modes, there is also a 95/95 probability/confidence that the peak kw/ft fuel rods will not exceed the melting temperature of UO_2, taken as 4900 of (unirradiated) and 4800 of at end of life. For these events, peak RCS pressure does not exceed 110 percent of the 2500 psia design pressure. The maximum average fuel pellet enthalpy was less than 200 cal/gm for all control rod ejection events, thus meeting the NRC criterion of less than 280 cal/gm.

2.15.3 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (SRP Section 15.4.7)

Although fuel assembly and burnable absorber loading errors are controlled by administrative procedures, fuel misloading errors, such as the inadvertent loading of one or more assemblies into improper positions, the loading of a fuel rod during manufacturing with one or more pellets of the wrong enrichment, the loading of a full assembly during manufacturing with pellets of the wrong enrichment, or incorrect placement of TPBARs were examined. DOE concludes, and the staff agrees, that should a loading error 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 then be evaluated. In the event that a loading error is not detectable by the instrumentation, any resulting fuel failures during normal operation would be minimal and offsite dose consequences would be a small fraction of 10 CFR Part 100 guidelines.

2.15.4 Steam Generator Tube Failure

DOE has evaluated the steam generator tube rupture (SGTR) analysis for the reference plant with regard to the effect of the TPBAR core design. DOE determined that the reactor core characteristics have only a minor effect on the SGTR analysis. Also, the core characteristics that have a minor effect on the SGTR are not significantly affected by the incorporation of a full complement of TPBARs. Therefore, DOE concludes, and the staff agrees, that the SGTR analysis of the reference plant FSAR is still bounding.

2.15.5 LOCA (SRP Section 15.6.5)

Large-break (LB) and small-break (SB) LOCAs were analyzed to estimate the response of TPBARs to the design-basis LOCAs and to assess the potential for interaction of the TPBARs with the LOCA transients. In its review, the staff also considered the applicability of the LOCA analysis methodologies to reactor cores with TPBARs.

The methodology used for LBLOCA analyses was the "1981 Evaluation Model" (SATAN and LOCBART codes) plus BASH. The SBLOCA analyses were performed with the Westinghouse SB Evaluation Model (NOTRUMP and SBLOCTA codes). Both of these evaluation models and their constituent codes have been generically approved for licensing-basis LOCA analyses in accordance with the provisions of 10 CFR 50.46 and Appendix K of 10 CFR Part 50. TPBAR assessments were performed with LOCTAJR with LOCA transient boundary conditions calculated using one of the approved codes. LOCTAJR is a version of the LOCTA code that has been modified to accommodate TPBAR material and geometry differences from those of rods or tubes in plants with conventional burnable absorbers. LOCTAJR has not been approved for nuclear plant licensing analyses. The staff is familiar with the approved versions of LOCTA from which the LOCTAJR code is derived. The LOCA conditions that TPBAR rods are calculated to experience are expected to be similar to those experienced by other burnable absorbers, thimble tubes, and core structural components. On this basis, the staff concludes that the approved methodologies mentioned above in conjunction with the LOCTAJR code are appropriate for the certification assessments for TPBAR.

LBLOCA and SBLOCA analyses were performed with the models discussed above, including the LOCTAJR code for cores with TPBARs, for demonstration operating conditions, and inputs and assumptions expected to bound actual plant inputs. Results indicate that TPBAR rods would behave acceptably under both LBLOCA and SBLOCA conditions. The staff concludes that the analyses are acceptable for the purposes of the proposed approval.

The staff concludes from its review that calculated TPBAR performance under LOCA conditions has demonstrated that TPBARs can be assessed with approved licensing LOCA models and can perform acceptably under LOCA conditions. However, the staff also concludes that, although the LOCTAJR code was appropriate for use in the demonstration analyses and assessments discussed herein, LOCTAJR was not reviewed for licensing use and should be reviewed by the staff for licensing applications and for its interface with the specific plant licensing LOCA models before it is used in specific plant licensing applications.

2.15.6 Radiological Consequences of Accidents

This section addresses the effect 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 LB LOCA and the fuel handling accident.

The radiological consequences of accidents are affected by operation with TPBARS in the core, primarily by the addition of the tritium from the TPBARs to the accident source term and the fact that the core source term is somewhat different than for operation without TPBARs.

The offsite radiological consequences for the reference plant design-basis accidents are evaluated against dose limits specified by the NRC for the thyroid and whole body (acute dose). For the control room dose evaluation, the evaluation is against a beta-skin dose limit in addition to the limits for the thyroid and whole body.

Loss-of-Coolant Accident Consequences

The radiological consequences of a LOCA are determined on the basis of 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. 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 2.91×10^{7} 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 2 hours or longer). With this assumption, most of the tritium will be in the sump solution and only about 3 percent of the tritium is available for leakage from the containment. When considering 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 plus 10 percent of the iodine in the non-flashed portion is assumed to become airborne. The iodine release from the ECCS leakage is reduced by filters with 90 percent removal efficiency. All of the tritium in the recirculation leakage is assumed to become airborne upon eventual evaporation of the water, and the release of tritium is not affected by filters.

The projected offsite doses are only slightly changed from those calculated for operation without TPBARs. The calculated site boundary dose (calculated for two hours immediately following an accident) for plant operation without TPBARs is 64.7 rem to the thyroid and 1.6 rem whole body. For plant operation with TPBARs, the thyroid dose is 61.4 rem and the whole body dose is 1.7 rem. These values are well below the NRC acceptance values of 300 rem for the thyroid dose and 25 rem for the whole-body dose.

The projected offsite dose to the low population zone (calculated for the assumed 30-day duration of the accident) for plant operation without TPBARs is 78.4 rem to the thyroid and 1.1 rem whole body. For plant operation with TPBARs, the thyroid dose is 74.4 rem and the whole-body dose is 1.2 rem. These values are well below the NRC acceptance values of 300 rem for the thyroid dose and 25 rem for the whole-body dose.

Fuel Handling Accident Consequences

For the reference plant, DOE assumed that in a fuel handling accident all the rods in a dropped assembly plus 20 percent of the rods in an affected assembly are damaged such that the activity in the fuel/cladding gap is released into the spent fuel pool water. Thus, it is appropriate to assume that the 24 TPBARs in the dropped fuel assembly plus 5 more in the affected 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 Ci) 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 spent fuel water pool.

At the water temperatures typically found in the spent fuel pool and refueling cavity, 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.

On this basis, the staff concludes that, although operation with TPBARs in the core does change the radioactive source term, there is a negligible effect on the ability of the plant to operate within regulatory requirements

Consequences of Other Design-Basis Accidents

Of the remaining design-basis events, only the RCP rotor seizure and the rod ejection accidents involve the release of radioactivity from the damaged fuel rods. It has been determined that there will be no damage to the TPBARs from these accidents. Thus, the only effect on the radiological consequences is from the changes in core source terms. These have been evaluated as resulting in a decrease in the projected thyroid dose of approximately 5 percent and an increase in the whole-body dose of approximately 10 percent. Thus, there is sufficient margin to the NRC's dose acceptance limits for the whole-body doses such that the identified increase in dose is not significant.

For accidents without fuel damage, such as an SGTR, main steamline break, small line break outside containment, and postulated failures of the gaseous or liquid waste processing systems, there is also no TPBAR damage. For these accidents, there is no effect on thyroid doses as a result of operation with TPBARs in the core, because reactor coolant iodine concentrations are limited by the technical specifications. The reactor coolant noble gas source terms are affected by operation with TPBAR. The changes in reactor coolant activity result in an increase in whole-body doses of about 6 percent. The whole-body doses remain within the NRC's dose acceptance limits.

Operation with TPBARs does result in an increase in the tritium released to the primary reactor coolant, but this increase is countered by an increased discharge of primary reactor coolant to prevent the tritium level in the coolant from exceeding the current operating level. Since reactor coolant tritium levels are not expected to increase, there is no adverse impact on the postulated liquid tank failure releasing waterborne activity to the groundwater.

On this basis, the staff concludes that, although operation with TPBARs in the core does change the radioactive source term, there is a negligible effect on the ability of the plant to operate within regulatory requirements

Impact of TPBAR Failure

In the event that a TPBAR suffers a cladding degradation late in the operating cycle, it is assumed that the full inventory of tritium could be released to the reactor coolant. A maximum inventory of 11,600 Ci of tritium in the rod would increase the reactor coolant concentration by 50 μCi/gm to 54 μCi/gm. If all of this tritium were to be released to the environment, it would have a negligible effect on the offsite doses, which would remain well within the NRC's acceptance limits.

In its evaluation, DOE did not include the whole body dose with the thyroid dose. The internal dose from tritium is a whole-body dose, as is documented in Federal Guidance Reports 11 and 12. This is not addressed explicitly in the SRP because, with the model source terms that have been used for typical power reactor operation, the dose from tritium is not significant. However, because the calculated doses are already well below the NRC acceptance criteria, the small percentage increase in the reactor coolant activity from the tritium would have a negligible effect on the offsite doses, which would continue to remain well below the NRC's acceptance criteria.

On this basis, the staff concludes that, although operation with TPBARs in the core does change the radioactive source term, there is a negligible effect on the ability of the plant to operate within regulatory requirements.

2.15.7 Anticipated Transients Without Scram (ATWS)

Introduction and Background

An anticipated transient without scram (ATWS) is an anticipated operational occurrence during which an automatic reactor scram is required, but fails to occur because some common mode fault in the reactor protection system. Although ATWS events are not considered to be design-basis accidents, a series of studies (WCAP-8330, NS-TNA-2182) on ATWS were conducted that showed that acceptable consequences will result, provided that the turbine is tripped and auxiliary feedwater flow is initiated within technical specifications time limits. The limiting criterion associated with the ATWS analyses for Westinghouse plants is that the maximum reactor coolant system (RCS) pressure does not exceed 3200 psig. This pressure corresponds to the ASME Code Service Level C stress limit in the most stressed limiting RCS component. The NRC ATWS rule (10 CFR 50.62), 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.

A major control component to an ATWS event is the net amount of reactivity feedback available at the time of the event. Because the ATWS event is by definition an event without the availability of the control rods, the core must rely entirely upon inherent feedback mechanisms, such as Doppler and moderator coefficients, to bring the core under control. The net effect of these coefficients is reflected in the calculation referred to as "unfavorable exposure time" (UET). The methodology behind this calculation follows.

Method of Analysis

The methodology underlining the determination of the UETs is described in detail in WCAP-11992. The TPC ATWS analysis focuses on two aspects of WCAP-11992, namely the UET and the critical trajectory methodologies. The staff's review of WCAP-11992 to evaluate this approach was restricted to the relevant sections (4.3.8, 4.6.8, and B.7.1) of the topical report. The critical trajectories are calculated loci of plant conditions (e.g., power-vs.-inlet temperature), which provide a peak pressure in the transient analysis of the limiting ATWS event at the specified limit of 3200 psig. The UET is the time during the cycle when reactivity feedback is insufficient to maintain pressure under 3200 psig for a given reactor state.

Calculating the UET for an ATWS event is a two-step process. First, the ATWS transient point kinetics information is converted into steady-state conditions (during peak ATWS pressure conditions, heatup is relatively slow so that steady-state analysis is acceptable), and the critical trajectories are determined. This information is later used to compare with cycle-specific core-condition evaluation calculations. Second, cycle-specific reference core calculations are performed with appropriate ATWS initial conditions of full power, rods out, equilibrium xenon, and 3200 psig RCS pressure. The reference core criticality is determined as a function of temperature. The reference core results are compared to the critical trajectories from the ATWS transient calculation results. This comparison provides cycle-specific design conditions, which would result in transient conditions exceeding 3200 psig. Calculations as a function of time in cycle, and thus as a function of moderator temperature coefficient (MTC), show the time during the cycle that the reference core design critical trajectories are greater than the ATWS critical transient trajectories. The period of time corresponding to the reference core trajectories exceeding the ATWS critical trajectories is referred to as the "unfavorable exposure time."

Calculation Results and Conclusion

DOE performed calculations to determine the effect of the TPBARs on the performance of the reference core with respect to the limiting ATWS event. Calculations of UET were performed for the TPCs. For comparison, UET calculations for a representative cycle of the reference core were also performed. The critical power trajectory assumed for these calculations was taken from WCAP-11992, and corresponds to the 100-percent auxiliary feed, loss of load ATWS with two PORVs available and no manual rod insertion. Two calculations were performed: Cycle 1 and the equilibrium cycle. The results of the UET calculations for both analyzed TPC cycle designs described in Section 2.4.3 of this report, were found to be substantially conservative when compared to the UET values for the reference core.

The UETs for the TPC designs are somewhat better than the current reference core because of the less positive MTC in the TPC designs. As discussed above, the large number of burnable absorbers (both IFBAs and TPBARs) 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 cores. This, in turn, reduces the critical powers and the UET. Reload cores that employ large numbers of TPBARs will always tend to have more-negative MTCs than standard reference plant 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 will lead to lower boron worths, more negative MTCs, and therefore, lower UETs.

The staff agrees with the partial ATWS analysis conducted and the results obtained by DOE. However, this concurrence pertains only to the TPC topical report. The staff concludes that licensees seeking to utilize a TPC must submit a plant-specific application containing a full ATWS analysis, conducted in accordance with NRC regulations and approved standards. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

2.16 Technical Specifications

During its review of the TPC topical report, the staff identified a number of potential technical specification changes that will be needed in order to support an application by licensees participating in DOE's program for the CLWR production of tritium. These include the following*:

TS 3.4.12, "Low Temperature Overpressure Protection (LTOP) System "
TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits"

In Section 2.5.2 of this report, the staff concludes that the Appendix G limits and the cold overpressure mitigation system (COMS) setpoints will probably be affected slightly by the incorporation of TPBARs. Therefore, for some older plants with P/T limits and COMS setpoints specified in the technical specifications, TS 3.4.12 and TS 3.4.3 may need to be revised. However, for plants with TSs that reference the PTLR for these values, only the PTLR would need to be revised on a plant-specific basis.

TS 3.7.17, "Spent Fuel Assembly Storage"
TS 4.3, "Design Features, Fuel Storage"

TS 3.7.17 contains a figure that defines acceptable combinations of fuel assembly burnup and initial U-235 enrichment for a particular fuel storage configuration. TS 4.3 describes in detail the allowed spent fuel storage configurations. As discussed in Section 2.9.2 of this report, because fuel storage racks and available storage space differ from plant to plant, the staff concludes that confirmation that a particular tritium production fuel management scheme will meet spent fuel storage k-eff limits will have to be provided on a plant-specific basis.

2.17 Quality Assurance

Section 2.17 of the TPC topical report describes the regulatory processes for reviewing the content of quality assurance (QA) programs applicable to the manufacture of the production core TPBAR components and to a plant that seeks to utilize a TPC. Neither the TPBAR manufacturer, vendors, or reference plant is identified in the report. Therefore, issues related to component procurement and fabrication must be addressed before a particular reactor facility can be authorized to irradiate TPBARs for the production of tritium. DOE submitted additional information related to quality assurance controls in letters dated July 30 and December 2, 1998. The staff has evaluated Sections 2.17.1 and 2.17.2 of the TPC topical report to determine the appropriateness of the regulatory processes for reviewing aspects of TPC QA programs.

The NRC has previously approved a license amendment for the irradiation of TPBAR lead test assemblies (LTAs) in the Watts Bar Nuclear Plant reactor core during Cycle 2. A summary of the staff's review of associated reactor licensee and Pacific Northwest National Laboratory (PNNL) submittals, including DOE technical report PNNL-11419, is documented in the safety evaluation supporting the Watts Bar LTA license amendment (NRC letter dated September 15,

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*The technical specification numbers referenced in this section refer to the numbers in NUREG-1431, Revision 1, "Standard Technical Specifications for Westinghouse Plants."

1997). The TPC topical report differs from these LTA-related submittals, principally in its focus on the reactor plant aspects affected by the greater number of TPBARs in the core.

The safety evaluation for the TPBAR LTAs has established that the TPBARs are a basic component as defined in 10 CFR Part 21 that, by definition, are designed and manufactured under a QA program that complies with the requirements of Appendix B to 10 CFR Part 50. The TPBAR LTAs are integral parts of the reactivity control system to keep the reactor core in a safe state, and are therefore, safety-related. TPBAR component safety functions and critical characteristics are identified in PNNL report TTQP-1-046. Critical characteristics are those important design, material, and performance characteristics necessary to provide reasonable assurance that the item will perform its intended safety function. Chapter 17 of the SRP applies to QA controls regarding TPBAR incorporation into the reference plant.

2.17.1 Quality Assurance During the Design and Construction Phases (SRP Section 17.1)

The acceptance criteria in SRP Section 17.1 are based on the relevant requirements of Appendices A and B of 10 CFR Part 50, 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. The acceptance criteria deal with the QA controls related to the 18 areas outlined in 10 CFR Part 50, Appendix B. Additional review guidance is provided by the regulatory guides referenced in SRP Section 17.1.

DOE has not yet selected the supplier for the fabrication of the production core TPBARs, and NRC review and inspection of supplier/vendor QA programs is not within the scope of this evaluation. Procurement processes performed on behalf of DOE for production core TPBAR components by contractors other than the production core TPBAR fabricator will also be subject to NRC review and inspection. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

2.17.2 Quality Assurance During the Operations Phase (SRP Section 17.2)

The staff reviews and evaluates a licensee’s description of its quality assurance program for the operational phase as described in the licensee’s final safety analysis report. The review addresses both the “offsite” and “onsite” quality assurance controls to be applied to those activities that may affect the quality of items important to safety during the operation, maintenance, and modification of a nuclear power plant The acceptance criteria in this SRP section are based on the relevant requirements of Appendices A and B of 10 CFR Part 50, 10 CFR 50.55a, and 10 CFR 50.34(b)(6)(ii) with emphasis on activities associated with the operations phase. Additional review guidance is provided by the regulatory guides referenced in SRP Section 17.2.

2.17.3 Reporting of Defects and Noncompliance (10 CFR Part 21)

10 CFR Part 21 requires suppliers of components to licensed facilities to notify the NRC, immediately following discovery of information reasonably indicating that the facility, activity, or basic component supplied to such a facility or activity (1) fails to comply with regulatory requirements or (2) contains defects that could create a substantial safety hazard. The production core TPBAR supplier will be required to comply with 10 CFR Part 21.

2.17.4 Conclusions

SRP Section 17.1 provides appropriate guidance for NRC review of quality assurance programs and activities related to the design and manufacture of TPBAR components. Inspection and review of the quality assurance programs controlling manufacturing and procurement processes will not be performed until a manufacturer has been selected by DOE.

SRP Section 17.2 provides appropriate guidance for NRC review of quality assurance programs and activities related to the operation of the reference plant. These QA programs and activities are plant-specific and therefore, must be addressed by licensees participating in DOE’s program for the CLWR production of tritium.

TPBARs are basic components as defined in 10 CFR Part 21 that, by definition, will have to be designed and manufactured under a QA program that complies with the requirements of Appendix B of 10 CFR Part 50.

2.18 Human Factors Engineering

Chapter 18 of the SRP deals with the adequacy of the control room design and the safety parameter display system, relative to human factors considerations. On the basis of a qualitative review of information presented by DOE to address the guidance of SRP Section 18.1, “Control Room,” and SRP Section 18.2, “Safety Parameter Display System,” the staff concludes that further evaluations of TPBAR effect in this area are not required. DOE concludes, and the staff agrees, that revisions to Chapter 18 of a typical SAR would not be required to accommodate irradiation of a tritium production core.

2.19 Summary and Conclusions

DOE has performed an assessment of the effect of using TPBARs in all available core locations on all aspects of a reference CLWR design, using the Commission's SRP as guidance. The staff concludes that, except where noted otherwise, DOE's TPC topical report acceptably assesses the impacts resulting from the incorporation of TPBARs into a plant of the reference design, and that it provides an appropriate design, methodology, and analysis for reference by licensees participating in DOE's program for the CLWR production of tritium. However, as discussed throughout this chapter, the TPC topical report identifies matters that must be addressed by licensees seeking to utilize a TPC because the actual design of an individual plant may not be bounded by the parameters of the reference plant. In addition, as summarized in Chapter 5 of this report, the staff has identified interface items that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to produce tritium for DOE. These items will be reviewed during the staff's safety evaluation of each such application.

3 TPC TPBAR EVALUATION

Chapter 3 of DOE's TPC topical report contains the various aspects of DOE's tritium-producing burnable absorber rod (TPBAR) evaluation, including the design requirements, mechanical and thermal hydraulic evaluations, performance nuclear design interfaces, materials considerations, test data, and surveillance.

3.1 Introduction

In Section 3.1 of its TPC topical report, DOE provides a description of the design, design basis, and the performance evaluation of 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 17x17 fuel array. DOE states that the TPBAR nuclear, mechanical, and thermal characteristics are comparable and compatible with those of conventional burnable absorber rods and, therefore, the TPBARs have only one reactor safety function: to perform their neutron absorption function in conjunction with the reactivity control system.

3.2 TPC TPBAR Design

In Sections 3.2.1 and 3.2.2 of the TPC topical report, DOE describes the design features, materials, and operation of the TPBAR in the TPC. The external dimensions and the austenitic stainless steel cladding of the TPBARs are similar to those of the standard Westinghouse burnable poison rod assembly (BPRA). The TPBARs use Li-6 instead of B-10 as the neutron-absorbing material. On the basis of the comparison of the TPBAR design parameters with those of the Westinghouse BPRA and the Westinghouse wet annular burnable absorber (WABA) given in Table 3.2-1 of the TPC topical report, the staff concludes that TPBARs are similar in form to BPRAs and WABAs. The TPBAR design is neutronically the same as that approved by the NRC and used in the lead test assembly (LTA) design for Watts Bar, except for the linear loading of Li-6 and the active absorber length. In the TPC design, the Li-6 linear loading is 0.030 gm/in., as opposed to 0.0247 gm/in. in the LTAs. The active length of the TPBARs is 127.5 in. and 128.5 in. for the first cycle and for the equilibrium cycle, respectively. The LTA active length was 142 in. The higher Li-6 loading in the shorter TPBARs allows for tritium production essentially equivalent to that in the LTAs. The shorter TPBAR active length will be used to enhance the axial power distribution shape in much the same way as part-length burnable absorbers are used in conventional plants.

The acceptability of the nuclear and thermal-hydraulic design of the TPBARs was evaluated in Sections 2.4.3 and 2.4.4 above.

3.3 Design Requirements

In Section 3.3 of its TPC topical report, DOE describes the design requirements for the TPC TPBARs, listing the functional requirements, the quantitative performance limits and requirements to be met by the TPBARs in the TPC, and the the significant TPBAR parameters used for the average rod, the peak rod, and enveloping generic design conditions. Significant TPC TPBAR generic design characteristics include:

- 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 17x17 VANTAGE+ fuel configuration.

DOE concluded that these conditions, in conjunction with the 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. The staff will confirm that these conditions and assumptions are valid for a particular core during its review of a plant-specific application for an amendment to the facility operating license authorizing irradiation of a TPC.

The reactor system pressures and temperatures to be used in the TPC TPBAR mechanical evaluation for Conditions I, II, III, and IV transients in Sections 2.3.2 and 2.15 are also listed. DOE states that the analysis assumptions 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.

DOE states that, for TPC TPBAR designs operating within the limits defined in the 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. DOE also provides a list of the types of evaluations that will be required for the TPBARs in a TPC. The staff will review these evaluations as part of its review of a plant-specific application for an amendment to the facility operating license authorizing irradiation of a TPC.

3.4 Mechanical Design Evaluation

In Section 3.4 of its TPC topical report, DOE evaluates the TPBAR design for credible combinations of thermal, neutronic, mechanical, and hydraulic interactions. DOE has evaluated the TPBAR for integrity of the pressure boundary (cladding and end plugs), and for absorber pellet stability.

3.4.1 Tritium Production and Design Life

The TPBAR is currently designed for a maximum production of 1.2 gm of tritium, while the peak tritium production per rod is expected to be 1.089 gm. The design life for mechanical evaluation is 520 effective full-power days (EFPDs), while the nominal life of the core is 494 EFPDs. With the 1.2-gm limitation and the design lifetime of 520 EFPDs, the TPBAR design evaluations demonstrate significant design margins. DOE states that the assumptions and design limits applied to the TPC TPBARs are more conservative than those applied to other commercial core component rods and fuel rods.

3.4.2 Cladding Design (Stress, Strain, and Stability)

The cladding and end plug are manufactured from 20-percent cold-worked Type 316 stainless steel (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 percent 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 stated in the Material Properties Handbook (MPH) and in the American Society for Testing and Materials (ASTM) 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. Normal operation is referred to as Condition I. Events postulated to occur often are referred to as Condition II events. Extremely low probability events, which have the potential to cause significant fuel damage, are classified as Condition III and IV events. 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 loss-of-coolant accident (LBLOCA) events, when failure of TPBARs is assumed to occur. The consequences of such failures are evaluated in Section 2.15.6 of this report.

The structural members (cladding and top and bottom end plugs) of the TPBAR were designed using stress and fatigue criteria as well as methodology consistent with the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code Section III as a guide. The external pressure criteria of the ASME 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 because the material properties of ASTM A 771 316 SS are not given in the ASME Code. The stress correlation is used to evaluate the discontinuity stress at the weld junction between the cladding and the end plug. The loads on TPBARs resulting from worst-case transient pressures, or from handling and shipping, are greater than those resulting from 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 of the TPC topical report. The design stresses were derived from the material properties of 20-percent cold worked 316 SS compiled in the MPH. The staff finds the safety margins reasonable and acceptable.

The results indicate that, except for the LBLOCA event in which 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 indicate the following:

- Critical buckling pressures were verified to be greater than the RCS design pressure of 2500 psia at the TPBAR design cladding 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 from 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.

- The irradiation creep and volumetric swelling strains are less than 0.2 percent. Nominal changes in cladding diameter dimensions from to irradiation are less than 0.0005 in. This is much less than the design limit of 1 percent on cladding strain.

LBLOCA and SBLOCA Considerations

At the LOCA temperature and pressure listed in Table 3.4-1 of the topical report, the TPBAR cladding stresses exceed design stresses. However, this does not necessarily represent cladding failure. A comparison of conservatively calculated rod cladding stresses with measured burst stresses of prototypical cladding indicates that the TPBARs are not expected to fail during an 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. The consequences of failure of the TPBAR during an LBLOCA event are discussed in Section 2.15.6 of this report.

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 subsection NG-3222.4(d) of the ASME Code, and therefore has the ability to withstand the cyclic service, and an analysis in accordance with subsection 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.

Cladding Collapse

The external pressure tests on the cladding 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 absorber pellets are made of lithium aluminate, a high temperature ceramic that is very stable at elevated temperatures. No densification or significant phase change of the absorber pellets is predicted over the range of temperatures encountered during Conditions I through IV. The absorber pellets are chemically stable and are non-reactive with other TPBAR components. The materials properties of the absorber pellets are discussed further in Section 3.8 of this report. Experience with irradiation of absorber pellets in the Advanced Test Reactor (ATR) has shown excellent stability up to a gas-volume ratio (GVR) of 239, with only minor microcracking. Because the maximum calculated GVR for the TPC is 207, absorber pellet disintegration, major cracking, and relocation is not expected.

3.4.4 Getter and Liner

The thermal and physical properties of the getters and liners are discussed in Section 3.8 of the TPC topical report. DOE has evaluated the effects of temperature, irradiation growth, and hydriding and has provided margin in the TPBAR design to account for these phenomena. The staff has reviewed DOE's treatment of these properties and concludes that the margins are adequate to allow for these components to meet their functional requirements.

3.4.5 Plenum Spring

The spring is made from 302 SS and is similar in design to springs used in the burnable poison rod assemblies (BPRAs) and fuel rods. The spring load stress has been established to be less than 60 percent 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 column buckling. No credit is taken for the spring in operational or reactor accident analysis. On the basis of the safety margin and satisfactory commercial reactor experience with this material, the staff concludes that the spring will provide the bearing load required for shipping and handling and is acceptable.

3.4.6 TPBAR Vibration and Wear Evaluation

Burnable poison (BP) and wet annular burnable absorber (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 that could result in wear damage of the rod cladding or the thimble tube. Experience with burnable absorber (BA) rods 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.

Burnable poison rod assemblies used in PWRs have not experienced failure from vibration fatigue. The fluid-induced TPBAR vibrations generate small cladding bending stresses. The maximum credible vibration stress as a result of the gap between the TPBAR and thimble guide tube was calculated to be an alternating stress of 2650 psi. This stress is significantly less than the endurance limit of 24,000 psi specified by the ASME Code. Therefore, the staff concurs with DOE's assessment that failure of a TPBAR due to vibration fatigue is not plausible.

On the basis of its review as discussed above, the staff concurs with the DOE assessment in the TPC topical report that the TPBAR cladding has adequate structural margin to failure, except during an LBLOCA event when it is assumed to fail.

3.5 <u>TPBAR Performance</u>

In Section 3.5 of its TPC topical report, DOE describes 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 the TPC topical report. A tritium release for the average rod of less than 1 curie per rod per year was assumed for the evaluations in Section 2.11 of this report.

DOE performed calculations for a TPC TPBAR with a design production of 1.2 g of tritium to verify that applicable TPBAR component design limits were met. Calculations for the core-average TPC TPBAR were also performed to verify that the tritium release to the coolant is less than 1 curie per rod per year, as assumed in analyses performed to support the radiological consequences evaluated in Section 2.11 of this report.

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. The TPC topical report contains an integrated calculation 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.

Based on its review of Section 3.5 of the TPC topical report, the staff finds that DOE has adequately addressed the issues of TPBAR performance and modeling in the following specific areas: TPBAR performance modeling (i.e., TPBAR temperatures, TPBAR internal helium pressure, TPBAR internal tritium pressure, pellet tritium release, getter tritium absorption, tritium permeation through the cladding, hydrogen ingress from the PWR coolant, and the estimated axial distribution of the getter loading in the design TPC TPBAR); tritium releases; performance during abnormal conditions; and failure limits.

The staff concurs with DOE's evaluation in the TPC topical report regarding TPBAR performance.

3.6 Thermal-Hydraulic Design Evaluation

In Section 3.6 of its TPC topical report, DOE evaluates the effect of the representative reactor core thermal hydraulic conditions on the function and integrity of the TPBARs. DOE used Westinghouse standard procedures 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. DOE concluded that TPBARs in the TPC generate 38 percent higher power than equivalent PYREX burnable absorber rods in the same reactor location, primarily due to the higher (n-α) reaction energy release in lithium-6 than in boron-10. Since the external features of both types of rods are almost identical, the guide thimble tube coolant flow remains unchanged. The staff's evaluation of the core thermal hydraulics is provided in greater detail in Section 2.4.4 of this report.

3.7 Nuclear Design Interfaces and Conditions

In Section 3.7 of its TPC topical report, DOE addresses nuclear design interfaces and conditions. DOE 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 for peaking factors, moderator coefficient, power tilt, boron concentration limits, etc. The higher reactivity worth of the lithium-6 in the TPC relative to boron-10 used to control core reactivity, and the current experience base in producing lithium-6 enriched aluminate, impose a tight lithium-6 loading tolerance of 0.030 g/inch ±4.2 percent on an individual pencil basis. DOE states that expected lot-to-lot variations in grams of lithium-6 per inch are not expected to be greater than 2.7 percent and will, therefore, be acceptable.

Because axial gaps between absorber pellets in a pellet stack or between pellets in adjacent pencils can cause power peaking in adjacent fuel rods, a nuclear requirement has been established that gaps between pellets shall cause power peaking of less than 3 percent for burnups below 10,000 MWD/MTU and less than 5 percent for burnups above 10,000 MWD/MTU. Additional margin is provided to accommodate power peaking in the second half of the cycle as a result of the flattening of the core power profile as the cycle progresses beyond the mid-point. DOE states that power peaking in the fuel rods adjacent to TPBARs with worst-case gaps will meet these functional requirements. This is discussed further in Section 2.4.3 of this report.

DOE has assumed that a maximum of two TPBARs fail or leak after start-up or at some time in the irradiation cycle, as discussed in Section 2.15.2 of this report. This is based on failures of commercial heterogeneous burnable absorber rods and WABA rods in PWRs, where 2 out of 29,700 rods and zero out of approximately 500,000 rods, respectively, failed during first cycle irradiation.

DOE has evaluated the potential impact of a TPC on RCS chemistry and concluded that, as the stainless steel exterior surface of the TPBARs is indistinguishable from other stainless steel components in the core, irradiation of the TPC will have no adverse impact on RCS chemistry. This is also discussed in Section 3.8, below.

DOE has described the consequences of potential handling damage resulting from refueling operations and during onsite fuel assembly movement and handling with TPBARs installed. If an irradiated TPBAR is breached as a result of mishandling in the spent fuel pool, only a small fraction of the tritium inventory would be released. The tritium in the open pores of the pellet (tens of Ci) will be released when water comes in contact with the pellet. Further release may occur gradually due to the limited leaching of the pellets and would provide adequate time to isolate the damaged TPBAR cluster to prevent further release into the pool. DOE did not address post-irradiation movement of the TPBARs outside of fuel assemblies. Therefore, the staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

Finally, DOE considered the impact of TPBAR absorber material relocation and the potential for RCS interaction with a water-logged rod. The reactivity changes that could potentially occur as a result of these events has been shown to be benign for credible gaps and with a water-logged rod. This is also discussed in Section 3.8, below.

3.8 Material Evaluation

TPBAR materials have been selected that are compatible with the range of reactor coolant system (RCS) operating conditions and with the irradiation environment in the TPC. The consequences of accidents were taken into account when selecting materials.

There were 32 LTA TPBARs that were fabricated and inserted into the Watts Bar Unit 1 reactor. The same materials used in the LTA are being used in the TPC TPBARs. The staff concluded that the materials engineering issues of the LTA had been acceptably addressed as stated in the staff's SER concerning LTAs containing TPBARs (NUREG-1607). The basis for this conclusion was that the LTAs met all of the ASME Code materials and design requirements for burnable poison rod assemblies. Furthermore, the staff noted that the materials used in the LTAs have many years of successful service in operating PWRs with no serious problems. The same materials are being used in the TPC TPBARs.

In the TPC topical report, DOE states that the TPBAR design consists of an American Iron and Steel Institute (AISI) 316 SS cladding that is coated on the interior surface with a permeation-resistant aluminized barrier coating. The TPBAR design for the TPC uses thin-walled annular lithium aluminate ($LiAlO_2$) pellets assembled into stacks extending over the full or partial length of the active core. The TPC TPBAR pellet stack length of $LiAlO_2$ pellets enriched in Li-6 is 128.5 in. for the reference equilibrium core and 127.5 in. for a first core, starting approximately 5.4 in. from the bottom end of the TPBAR and approximately 7.5 in. above the bottom of the active fuel pellets in the adjacent fuel rods. Above and below the Li-6 enriched $LiAlO_2$ pellets are short stacks of pellets fabricated from depleted lithium aluminate. These short pellet stacks are used to position the enriched pellet column within the core. The pellets are packaged into 12 stacks of components referred to as "pencils." The getter tube surrounds the absorber pellets and is composed of nickel-plated Zircaloy-4.

3.8.1 Materials Specification

DOE states that the materials of construction for the TPBARs were chosen on the basis of successful experience in commercial service, in-reactor and ex-reactor testing programs, and for compatibility with other reactor internals, fuel assemblies, the reactor coolant system, fuel pool equipment, and fuel pool cooling systems.

DOE states that the materials of construction are procured and fabricated using ASTM standards for the 316 SS cladding and end plug, the Zircaloy-4 liner and getter, the nickel plating of the getters, and the plenum spring. Certified material test reports (CMTRs) are prepared for all TPBAR components.

A number of non-destructive (NDE) techniques are used during the fabrication of the TPBARs, including visual inspection, ultrasonics, eddy current, radiography, and helium leak testing. When appropriate, the NDE techniques conform to commercial standards.

The inner surface of the 316 SS cladding is coated with a permeation-resistant aluminum barrier coating. After the barrier coating is applied, NDEs are performed to verify coating thickness, coating integrity, and coating consistency.

The staff concludes that the materials of construction for the TPBARs are acceptable on the basis of years of successful service in operating PWRs. The most commonly used austenitic stainless steel in the nuclear industry is 304 SS. This material has a nominal composition of 19-percent chromium, 9-percent nickel, 1-percent manganese, and up to 0.08-percent carbon. This material is used for cladding in dummy fuel pins that shield reactor vessel walls from radiation damage, for active fuel pins, for reactor vessel cladding, for primary coolant system piping and valves, and as fuel cladding in four domestic PWRs. A significant increase in corrosion resistance is achieved by using 316 SS instead of 304 SS. Type 316 SS has a nominal composition of 17-percent chromium, 12-percent nickel, 2.5-percent molybdenum, 1-percent manganese, and up to 0.08-percent carbon. The molybdenum addition gives the 316 SS improved corrosion resistance and also provides higher resistance to creep, greater stress-to-rupture, and greater tensile strength at elevated temperatures. Type 316 SS has been used in nuclear piping, pumps, valves, and previous tritium target cladding.

The standards used for the construction of the TPBARS are primarily ASTM standards that were developed specifically for the construction of components to be used in nuclear applications. The staff concluded in the safety evaluation for the LTAs that the reliance on ASTM standards for the purchase of cladding satisfies the requirements of 10 CFR Part 50, Appendix B.

The 316 SS cladding is purchased and constructed in accordance with ASTM A 771, which will meet the requirements of Appendix B to 10 CFR Part 50 and the provisions contained in the licensee and supplier NRC-approved Quality assurance (QA) program descriptions as provided for in Section 2.17.1 of DOE's topical report.

3.8.2 Materials Properties

The DOE TPC topical report states that the properties of the materials used in the TPBAR are compiled in the Materials Properties Handbook (MPH) in the same manner as was done for the LTA.

The 316 SS cladding aluminum barrier coating (tritium barrier) increases the amount of tritium retained inside of the TPBAR. The barrier material to be used in these TPBARs has been selected through a barrier selection process. The barrier selected has resistance to corrosion, irradiation, and mechanical stresses that are similar to stainless steel cladding.

The ASTM A 771 316 SS cladding is 20-percent cold-worked, resulting in higher allowable stress between 100 °F and 850 °F when compared to 304 SS. This also results in a higher fatigue endurance limit that is unlikely to be exceeded during a normal operating cycle.

The effect of barrier application on the strength of the cladding has been examined and the effect is small. The small reduction in strength has been included in the design analysis for the TPBAR.

The cladding was tested by Pacific Northwest National Laboratory (PNNL) for collapse at the start of the cycle where the pressure in the reactor coolant boundary (RCB) is much higher than the internal pressure in the TPBAR. As tritium and helium are produced, the pressure inside the cladding increases and the tendency to collapse decreases. The cladding did not collapse or significantly deform when the test pressure exceeded the RCB pressure during normal operation.

The mechanical performance of the absorber pellets is controlled by mechanical strength, density, irradiation, swelling, and gas release. The strength of the pellets is sufficient to handle shipping and handling loads. Post-irradiation examinations of pellets indicate minor cracking but no evidence of loss of pellet integrity from irradiation. Swelling from irradiation to the design levels was insignificant. Gas retention in the pellets depends on production rate and temperature. Most of the helium generated in the pellets is released from the pellets as is most of the tritium. The tritium getters undergo a slight swelling due to hydriding and irradiation-induced growth.

The staff concludes that DOE has demonstrated through analysis of experimental data that the design factors of safety required to avoid cladding collapse have been met. This, along with DOE's operating experience with TPBARs, gives adequate assurance that the cladding will remain free-standing and will not collapse because of external pressure or creep for the design life. The staff further concludes that DOE has presented analyses and operating experience that give reasonable assurance that the absorber pellets will maintain their integrity during tritium production.

The TPC topical report states that during an LBLOCA, the component peak temperatures will be below the melting temperature of the pellets, the cladding, and the zirconium. However, the burst stresses for the 316 SS cladding will be exceeded, as was the case for the LTAs.

The staff finds that the material properties are acceptable because they equal or exceed the properties of materials currently used in operating reactors. In particular, the cold-worked 316 SS is stronger and has better fatigue life than the commonly used 304 SS, which has years of successful service in operating PWRs.

Furthermore, the staff finds that DOE's analysis, experimental data, and operating experience offer reasonable assurance that the cladding will not be affected for Conditions I, II and III. On the basis of cladding stress calculations, DOE states that cladding breach is not expected during an SBLOCA. However, because high cladding pressures occur at elevated temperatures during an LBLOCA, it is likely that the TPBAR cladding would fail under postulated accident conditions, and DOE's experimental data indicate that the cladding is expected to fail during an LBLOCA (Condition VI). Burst testing of specimens indicates that the cladding will burst at about 1500 °F (815.5 °C) and 5230 psia (36.1 MPa), compared to a predicted LBLOCA temperature of 2200 °F (1204 °C) with a differential pressure across the cladding that would exceed 5230 psia (36.1 MPa). Section 2.15.6 of this report addresses the impacts of a TPBAR rupture.

3.8.3 Material Compatibilities for Normal and Accident Conditions

Operating experience in PWR and boiling-water reactor (BWR) plants with stainless steel cladded fuel rods, control rods, and structural components indicates that the uniform corrosion rate of 304 SS is small, less than 0.1 mil per year (mpy). Type 316 SS has higher resistance to general corrosion, pitting corrosion, transgranular stress corrosion cracking, and intergranular stress corrosion cracking than 304 SS. 316 SS also has greater strength and resistance to creep than Type 304 SS. A corrosion allowance for the 316 SS cladding incorporated in the design is about 0.3 mil.

Experience in PWRs has shown that there is insufficient oxygen present to cause stress corrosion cracking (SCC) in austenitic stainless steel in PWR coolant. Furthermore, SCC will not occur during Condition III events because the cladding tensile stress is low and the duration of these events is short.

The TPC topical report states that the TPBAR internal and cladding materials are compatible with each other except during an LBLOCA. The TPBAR internal components are mechanically, chemically, and metallurgically compatible during Conditions I, II, and III. Only minor metallurgical interactions occur during Condition IV, except during the maximum LBLOCA. In-reactor and ex-reactor test results indicate that the aluminide barrier coating will not peel or blister during Conditions I, II, III, and IV.

The staff finds that DOE has provided reasonable assurance that the stainless steel cladding will have a low uniform corrosion rate because years of experience in operating PWRs have shown that 304 SS has a low uniform corrosion rate, and 316 SS with its higher nickel, chromium, and molybdenum contents, will have an even lower corrosion rate. Also, the addition of the molybdenum reduces significantly the probability of pitting corrosion or crevice corrosion. DOE states that the TPBARs are designed to be free of crevices, so crevice corrosion should not be a concern. Finally, transgranular stress corrosion cracking and intergranular stress corrosion cracking have not been observed for Type 304 SS in operating PWRs with hydrogen water chemistry due to the low oxygen content in the primary coolant. There have been reported instances of Type 304 SS stress corrosion cracking during storage in borated water with no hydrogen water chemistry. Type 316 SS is about four times more resistant to transgranular stress corrosion cracking and intergranular stress corrosion cracking than Type 304 SS.

In the analysis criteria for the TPBARs, the TPC topical report assumes that there will be two leaking TPBARs per production core during Conditions I and II. Reactor coolant water may enter the TPBARs and could dissolve some of the aluminide barrier, thus releasing a small quantity of Al_2O_3, water-soluble $AlCl_3$, and suspended solids. The releases from 3400 breached TPBARs would not exceed the Electric Power Research Institute (EPRI) guidelines for PWR water.

During Condition IV, the onset of getter melting will occur at the maximum temperatures reached during an LBLOCA. If a TPBAR ruptures during an LBLOCA, steam can react with the TPBAR internals, but only on a limited basis. There is no driving force for steam to enter the TPBAR through the narrow crack that is expected to form.

Although the $LiAl0_2$ absorber pellets may have limited leaching of lithium in water, as a result of the stability of the absorber pellets and the confinement of the pellets in the getter and liner tubes within the cladding, the possibility of pellet dissolution is extremely remote.

The staff finds it unlikely that the approximately 3400 TPBARs would be breached at the same time, and even if they were, the primary coolant would still meet EPRI guidelines for PWR primary coolant. The EPRI guidelines are based on controlling contaminants to levels that would not degrade materials in contact with the primary coolant. The staff also finds it unlikely that primary coolant will come into contact with the pellets since there are multiple barriers between the primary coolant and the pellets. Also, even if the primary coolant were to come into contact with the pellets, the pellets are not soluble in water.

3.9 TPBAR Irradiation Tests and Test Data Summary

Over the past 30 years, the Department of Energy has developed a significant amount of research information relative to the design, manufacture and irradiation performance of pressurized water reactor tritium target rods. Irradiation tests have been conducted on tritium target materials and design configurations, including some configurations similar to, but not identical to the current TPBAR design. Extensive ex-reactor testing on unirradiated (as-fabricated) components was also carried out, which supplemented and supported the in-reactor tests and post-irradiation examination results. Non-destructive examinations (visual, neutron radiography and external dimensions) of the irradiated test materials have confirmed that the cladding and internal components maintained their configuration under irradiation. Destructive examinations of irradiated TPBARS have also been performed.

DOE’s 1996 report on TPBAR LTAs (PNNL-11419) described a proposal by DOE to irradiate LTAs containing TPBARs in TVA’s Watts Bar reactor. The LTAs (4 assemblies containing 8 TPBARs each) were installed in the reactor core in September 1997, as part of the first Watts Bar refueling outage. The reactor operated successfully with the LTA TPBARs during Cycle 2 between October 1997 and February 1999 before shutting down for refueling. The LTAs will be examined by DOE after they are removed from the reactor. Current plans are to ship the LTAs to Idaho National Environmental Engineering Laboratory (INEEL) for radiography and non-destructive examination and then to PNNL for post-irradiation examination (PIE) (destructive and non-destructive).

The staff has reviewed the design of the TPBARs and the associated test results and concludes that DOE has presented sufficient analyses, test data, and operating experience data to give reasonable assurance that the TPC TPBARs will be compatible with the environment in the core of a PWR. Information from the LTA demonstration is expected to be provided to the NRC staff for their review prior to issuing a plant-specific license amendment for irradiating production quantities of TPBARs.

3.10 Planned Post-Irradiation Examinations for the LTA TPBARs

Empirical data for the TPBAR were generated in the Advanced Test Reactor (ATR). The irradiation of the 32 LTA TPBARs at Watts Bar Nuclear Plant will produce the first data generated in an operating commercial nuclear power plant. Nondestructive examination (NDE) and destructive PIE of the LTA TPBARs will be used to confirm the TPBAR design methodology.

The TPC topical report states that after irradiation, the PIE will be used to compare the results obtained in the ATR with the LTA results obtained at Watts Bar. The PIE will attempt to corroborate the general functional requirements for the production TPBAR design and will provide information to test the analytical models and the modeling assumptions.

3.10.1 Currently Planned PIE Strategy and Logistics

Following irradiation at Watts Bar, all 32 LTA TPBARs will be subjected to NDE. Although the intensity of the examinations may vary on the basis of initial findings, in general, each TPBAR will be subjected to visual examination, photography, gamma scanning, neutron radiography, diameter profilometry, and rod puncture/gas analysis. In addition, selected rods will be sectioned for destructive and quantitative examinations and testing. The preliminary plan is to examine four LTA TPBARs destructively.

3.10.2 Nondestructive Examinations

Nondestructive examinations of the LTA TPBARs irradiated at Watts Bar will take place at INEEL in Idaho Falls, Idaho. Rods will be visually examined over their full length in at least two orthogonal orientations with documentation by photo or video. Unusual features, such as excessive wear, scratches, pits, dimples, or patterned corrosion layers will be identified. The rod diameter and length will be measured. The rods will be neutron-radiographed over their full length to determine the location and physical state and location of pencils and absorber pellet columns. Rods will be gamma scanned to qualitatively assess rod-rod variation in cladding activation and neutron fluence and to characterize axial distribution of activation. The rods will be punctured and the plenum gas quantity measured. Then, the rods will be back filled with noble gas to determine the void volume.

The staff finds that the NDEs are unique to the TPBARs and exceed the requirements for burnable poison rod assemblies. The NDE techniques to be employed will adequately characterize the TPBARs.

The staff concludes that the TPBARs are being classified as safety related and will be fabricated to the design criteria of Section III of the ASME Code. The NDE techniques and applicable standards should conform to the requirements of Section III. However, since the TPBARs are not ASME Code components, it will not be necessary to request relief from the requirements of the code pursuant to 10 CFR 50.55a.

3.10.3 Destructive Examinations and Tests

Destructive examination of the TPBARs irradiated at Watts Bar will be conducted at DOE's Pacific Northwest National Laboratory in Richland, Washington. Destructive examinations will vary from rod to rod, but will include characterization of axial distribution and level of lithium burnup in the pellets; axial distribution of tritium and protium concentrations in the components, distribution of helium concentration in the pellets; metallurgical and microscopic evaluation of the getters and cladding (including the barrier layer); dimensional and physical characterization of cladding, getter, pellets, and liners; measurement of gettering rate; and measurement of the tritium permeation rate.

3.10.4 TPBAR Functional Requirements and Planned PIE

The PIE results will characterize the performance of TPBARs exposed to a full PWR operating cycle. The PIE results can be used to compare the LTA's TPBAR design with the TPC design and to assess the adequacy of the design. The information obtained will provide experimental verification of estimates of the amount of tritium produced, the tritium permeation release rate, the physical and dimensional stability of the $LiAlO_2$ absorber column, and the cladding structural integrity.

The staff finds that the PIE results will adequately baseline the performance of the TPBARs.

3.11 <u>TPBAR Surveillance</u>

In Section 3.11 of its TPC topical report, DOE proposes a surveillance program to confirm the satisfactory operation of the host plant with a large number of TPBARs. The surveillance program will be implemented during the first and second cycles of operation with production quantities of TPBARs in the core. A representative number of rods will be inspected after the first and second cycles. The surveillance program will include monitoring of coolant activity, periodic reviews of critical boron and in-core instrumentation measurements to compare the reactivity and power distribution of the production core with predictions, and post-irradiation examination of the TPBARs. The post-irradiation examinations will include visual inspection of 5 to 10 percent of the TPBAR cluster assemblies for evidence of wear or corrosion, loss of structural integrity, or other anomalies, and shipment of irradiated TPBARs to a DOE-specified site for additional post-irradiation examinations.

The staff concludes that the TPBAR surveillance program gives reasonable assurance that adverse impacts on core operation will be detected by the monitoring of coolant activity and the periodic reviews of critical boron and in-core instrumentation measurements. Furthermore, the staff finds that PIE by visual examination of TPC TPBARs in the spent fuel pool provides additional assurance that wear or corrosion, loss of structural integrity, or other anomalies will be identified. Finally, the staff finds that the since PIE will be conducted after each operating cycle, licensee surveillance will identify evidence of problems with the TPBARs in a reasonable time period.

3.12 <u>Summary and Conclusions</u>

The TPBAR as evaluated meets accepted and conservative criteria for materials selection and design for a core component in a Westinghouse 17x17 type fuel assembly inserted in a PWR. The TPBARs absorb neutrons as part of the fuel cycle reactivity control, and produce and retain tritium.

The 316 SS cladding is purchased and constructed according to the standards in ASTM A 771, which meets the requirements of Appendix B to 10 CFR Part 50 and the provisions contained in the licensee and supplier NRC-approved QA program descriptions as provided in Section 2.17.1 of the DOE topical report.

The TPBARs perform their function with acceptable margin to failure during normal operation and in conjunction with design-basis accidents with the exception of an LBLOCA (Condition IV). TPBARs use materials with known and predictable characteristics in reactor performance and are compatible with the reactor coolant system.

The staff has reviewed the materials used in the TPBAR and agrees that they are adequate for the TPC. On the basis of experimental results and operating experience in nuclear reactors, the staff finds that the materials in the TPBAR will not be affected by the environment and will not be adversely affected during Conditions I – IV, with the exception of an LBLOCA (Condition IV). The consequences of cladding failure would be inconsequential, as discussed in Section 2.15.6 of this report.

Furthermore, although the proposed NDE methods do not conform to the requirements of Section III of the ASME Code, the TPBARs are not ASME Code components and, therefore, it will not be necessary to request relief from the requirements of the code pursuant to 10 CFR 50.55a.

4 DOE CONCLUSIONS AND NO SIGNIFICANT HAZARDS EVALUATION

Section 4 of DOE's TPC topical report presents DOE's conclusions regarding the impact that the TPC would have on a typical pressurized-water reactor (PWR) (and vice versa) and discusses a determination that may be reached by the Commission in accordance with the provisions of 10 CFR 50.92(c) that a proposed amendment to an operating license does not involve a significant hazards consideration.

The staff's review conclusions regarding the TPC topical report are presented in Section 5 of this report.

4.1 DOE Conclusions

Section 4.1 of the DOE topical report discussed the evaluations and analyses that were performed to design a TPC that incorporates a full complement of tritium-producing burnable absorber rods (TPBARs) in order to maximize tritium production, while maintaining, to the extent possible, the same key accident analysis input parameters as currently exist. The topical report also evaluates the impact of the TPC on a representative plant design (the reference plant), using the Standard Review Plan as a guide. In addition, the topical report evaluated the impact of the reference plant parameters on the TPBAR design.

4.2 Determination of No Significant Hazards Consideration

Section 4.2 of the TPC topical report discussed how standard Westinghouse reload analysis methodology was used to determine whether any of the key safety analysis input parameters for the reference plant are affected by the TPC. DOE also performed accident analysis if there were any changes to the reactor plant systems or to the control or protection systems, whether as a result of the TPC or the associated reload core design.

A determination may be reached by the Commission in accordance with the provisions of 10 CFR 50.92(c) that a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not

(1) involve a significant increase in the probability or consequences of an accident previously evaluated or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated or

(3) involve a significant reduction in a margin of safety.

Before the Commission can reach a determination regarding a no significant hazards consideration on any license amendment request, the staff must complete its plant-specific review. Accordingly, a no significant hazards determination cannot be made at this time, based wholly on the TPC topical report.

5 SUMMARY AND CONCLUSIONS

The staff has reviewed the Department of Energy's (DOE's) topical report on the tritium production core and the related supporting information. Many technical issues have been satisfactorily addressed in the DOE topical report, as documented in this safety evaluation.

However, as discussed throughout this report, the TPC topical report identifies matters that must be addressed by licensees seeking to utilize a TPC because the actual design of an individual plant may not be bounded by the parameters of the reference plant. In addition, during its review, the staff identified certain interface issues that will require changes to the plant safety analysis report and that must be reviewed by the staff before the staff can determine the acceptability of irradiating a full-core load of tritium-producing burnable absorber rods (TPBARs) in any particular reactor facility. Therefore, the staff concludes that should any licensee wish to undertake irradiation of TPBARs, it must first submit an application for an amendment to the individual facility operating license for authorization to conduct such irradiation. Such application must address the plant-specific interface issues identified in Section 5.1 of this report and must include the necessary changes to the technical specifications located in Appendix A to the operating license.

5.1 Plant-Specific Interface Issues

During its review of the DOE tritium production core (TPC) topical report, the staff determined that there are certain plant-specific interface issues for which a licensee must submit additional information and analyses in support of a plant-specific amendment to the facility operating license for authorization to operate a tritium production core. These issues are listed below, along with the section(s) of this report in which each is discussed.

(1) handling of TPBARs (1.3, 2.9.2, 3.7)

(2) procurement and fabrication issues (1.3, 2.17.1)

(3) compliance with DNB criterion (2.4.4)

(4) reactor vessel integrity analysis (Appendices G and H to 10 CFR Part 50 and 10 CFR 50.61) (2.5.3)

(5) control room habitability systems (2.6.1)

(6) specific assessment of hydrogen source and timing or recombiner operation (2.6.2)

(7) light-load handling system (2.9.1)

(8) station service water system (2.9.1)

(9) ultimate heat sink (2.9.1)

(10) new and spent fuel storage (2.9.2)

(11) spent fuel pool cooling and cleanup system (2.9.3)

(12) component cooling water system (2.9.4)

(13) demineralized water makeup system (2.9.5)

(14) liquid waste management system (2.11.2)

(15) process and effluent radiological monitoring and sampling system (2.11.5)

(16) use of LOCTAJR code for LOCA analyses (2.15.5)

(17) ATWS analysis (2.15.7)

5.2 Effect on Plant Technical Specifications

During its review of the DOE TPC topical report, the staff determined that a facility undertaking irradiation of a tritium production core will require changes to the technical specifications contained in Appendix A of any facility operating license. These potential changes must be submitted to the staff for review and approval as part of an application for an amendment to the facility operating license that would authorize operation with the tritium production core. These changes are listed below, along with the section(s) of this report in which each is discussed:

1. TS 3.4.3 - RCS Pressure and Temperature (P/T) Limits (2.5.2, 2.16)
2. TS 3.4.12 - Low-Temperature Overpressure Protection (LTOP) System (2.5.2, 2.16)
3. TS 3.7.17 - Spent Fuel Assembly Storage (2.9.2, 2.16)
4. TS 4.3 - Design Features, Fuel Storage (2.9.2, 2.16)

APPENDIX A

TRITIUM-PRODUCING BURNABLE ABSORBER ROD (TPBAR) TRITIUM PRODUCTION CORE (TPC) FAILURE MODES AND EFFECTS ANALYSIS (FMEA)

DOE performed a failure modes and effects analysis (FMEA) to evaluate the potential consequences of the failure of the TPBARs and each of the TPBAR components. This analysis appeared as Appendix A to the DOE TPC topical report. Five failure modes were identified that had the potential to result in the inability of the TPBARs to perform their safety function. These five potential failure modes are (1) misplacement of multiple fuel assemblies in the core, (2) multiple TPBARs not loaded, (3) missing multiple pencils of absorber pellets, (4) lithium loading error affecting multiple TPBARs, and (5) 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. In addition, errors sufficiently large so that fuel design limits are exceeded would be readily detected by technical specification requirements for startup and flux map surveillance. Therefore, the staff concurs that the risks associated with these potential failure modes are acceptably small.

The failure of a TPBAR is less likely than the failure of a fuel rod because of the protected location of the TPBARs within guide thimbles. Their protected location also tends to preclude the interference of TPBARs with adjacent fuel rods.

APPENDIX B

CHRONOLOGY OF CORRESPONDENCE

This appendix contains a chronological listing of correspondence between the Nuclear Regulatory Commission (NRC) and the Department of Energy (DOE) and other correspondence related to DOE's topical report on the tritium production core. All documents, with the exception of certain enclosures to correspondence marked with an asterisk (*) (denoting "confidential restricted data") have been placed in the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, D.C., under Project No. 697.

July 30, 1998	Letter from S. M. Sohinki (DOE) to Document Control Desk (NRC), submitting Westinghouse topical report "Tritium Production Core (TPC) Topical Report," June 1998, NDP-98-153* and Westinghouse topical report "Tritium Production Core (TPC) Topical Report (Unclassified, Non-Proprietary)," July 1998, NDP-98-181.
July 30, 1998	Letter from S. M. Sohinki (DOE) to Document Control Desk (NRC), submitting QA responses to staff questions at December 17, 1997, meeting on scope and goals of tritium production core topical report.
August 6, 1998	Notice of public meeting on August 18, 1998, between NRC staff and DOE to discuss DOE's topical report on the tritium production core.
August 11, 1998	Letter from T. H. Essig (NRC) to S. M. Sohinki (DOE), transmitting *Federal Register* notice of receipt of DOE topical report on tritium production core.
August 26, 1998	Summary of meeting held on August 18, 1998, between NRC staff and DOE concerning tritium production core topical report.
September 11, 1998	Letter from T. H. Essig (NRC) to S. M. Sohinki (DOE), transmitting staff's schedule for NRC staff review of DOE topical report on tritium production core.
September 28, 1998	Notice of public meeting on October 8, 1998, between NRC staff and DOE concerning DOE's tritium program.
September 29, 1998	Letter from T. H. Essig (NRC) to S. M. Sohinki (DOE), transmitting staff's requests for additional information regarding DOE's topical report on tritium production core.
October 15, 1998	Letter from T. H. Essig (NRC) to S. M. Sohinki (DOE), transmitting staff's supplemental requests for additional information regarding DOE's topical report on tritium production core.

October 26, 1998	Letter from S. M. Sohinki (DOE) to Document Control Desk (NRC), providing correction to identified failure rate for commercial burnable rods.
November 5, 1998	Summary of public meeting held on October 8, 1998, between NRC staff and DOE to discuss staff's requests for additional information concerning tritium production core topical report.
December 2, 1998	Letter from S. M. Sohinki (DOE) to Document Control Desk (NRC), submitting responses to staff's requests for additional information regarding the tritium production core topical report.
January 13, 1999	Letter from S. M. Sohinki (DOE) to Document Control Desk (NRC), submitting supplemental responses to staff's requests for additional information regarding the tritium production core topical report.
February 10, 1999	Letter from S.M. Sohinki (DOE) to Document Control Desk (NRC), submitting Westinghouse topical report "Tritium Production Core (TPC) Topical Report," February 5, 1999, NDP-98-153 Revision 1* and Westinghouse topical report "Tritium Production Core (TPC) Topical Report (Unclassified, Non-Proprietary," February 8, 1999, NDP-98-181 Revision 1.
March 5, 1999	Letter from Thomas O. Martin (NRC) to S. M. Sohinki (DOE), concerning cognizant security agency responsibilities for facility security clearances.

APPENDIX C

REFERENCES

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Federal Guidance Report 12, “External Exposures to Radionuclides in Air, Water, and Soil.” U.S. Environmental Protection Agency. 1993.

Letter, September 15, 1997, from R.E. Martin (NRC) to O.D Kingsley (TVA). “Issuance of Amendment 8 to Facility License No. NPF-90, Watts Bar Nuclear Plant, Unit 1, on Tritium Producing Burnable Absorber Rod Lead Test Assemblies.”

NS-TMA-2182. Letter, December 1979, from T.M. Anderson (Westinghouse) to S.H. Hanauer (NRC). “ATWS Submittal.”

NUREG-0800. U.S. Nuclear Regulatory Commission. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." June 1987.

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